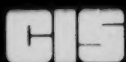


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Volume II-- Working Papers: Nuclear Power in an Age of Uncertainty

Part 1: Reliability of Nuclear Power Plant Hardware: Past Performance and Future Trends; Reactor Quality; Factors Affecting Managerial and Institutional Competence of Utility Nuclear Operations; Survey and Assessment of the Technological Options Available to the Nuclear Industry in the 1980 to 2000 Period; Nuclear Powerplant Size: Dependent Characteristics.

November 1984

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RELIABILITY OF NUCLEAR POWER PLANT HARDWARE

Past Performance and Future Trends

Prepared for:

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Past Performance and Future Trends

January 15, 1983

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One important consideration for the assessment of the future of nuclear power is the expected reliability of the reactor hardware itself. This report presents the results of a study of the past performance of reactor hardware and present areas of concern, together with an estimate of future trends based on our present knowledge. The study of past performance was conducted at two levels. The first looked at performance in the aggregate--the overall hardware performance and reliability as measured by the conventional indices of capacity factor, plant availability and operating and maintenance (O&M) costs. These are important bottom-line measures of how power-generating alternatives affect the cost of electricity and thus will influence future choices.

The second level examined the specific contributors to the loss of plant availability and reviewed their varied causes and the methods used to remedy them. In this manner, conclusions are reached concerning the nuclear industry's ability to deal with future hardware problems as they arise, as well as the trends for the occurrence of future problems.

To arrive at the conclusions about future trends, a study was made of current hardware issues of concern, both unresolved operational hardware problems, as well as possible future hardware problems resulting from regulatory concerns. Finally, the methods by which potential hardware problems are identified have been discussed and estimates made of the likelihood that future problems will be identified before they develop into major causes of nuclear plant downtime and extensive and expensive repairs.

Throughout this paper on hardware reliability, the perspective is maintained that the appropriate measures of performance for

nuclear power plant hardware are availability, capacity factor, O&M costs, and ultimately, the cost of electricity. It is unfortunate that, for nuclear power, the normal problems of hardware reliability are compounded by great public concern and media exploitation, as well as by increased visibility engendered by extreme over-regulation. These institutional issues are addressed elsewhere. The following sections present the results of this study: Section 2.0, Overview and Summary; Section 3.0, Potential Hardware Reliability Issues of Concern; Section 4.0, Future Trends in Reactor Hardware Reliability; and Appendix A, Review of Past and Present Hardware Reliability Problems.

Trends in Hardware Reliability

There is in some quarters a perception of nuclear power reactor technology that it is unreliable, unforgiving and more costly than its alternatives. A careful examination of plant performance data since the mid to late 1960's shows this not to be an accurate assessment. Examination of data from the National Electric Reliability Council for the period 1969-1978⁽¹⁾, as given in Figure 1, shows that to within a few percentage points, the availability* of the average nuclear power plant is indistinguishable (about 2% greater) from that of the average large sized coal-fired power plant. With the exception of relatively inexpensive hydro power, coal and nuclear power are the present preferred choices for base-load electric power generation, and therefore the two provide the most meaningful comparison.

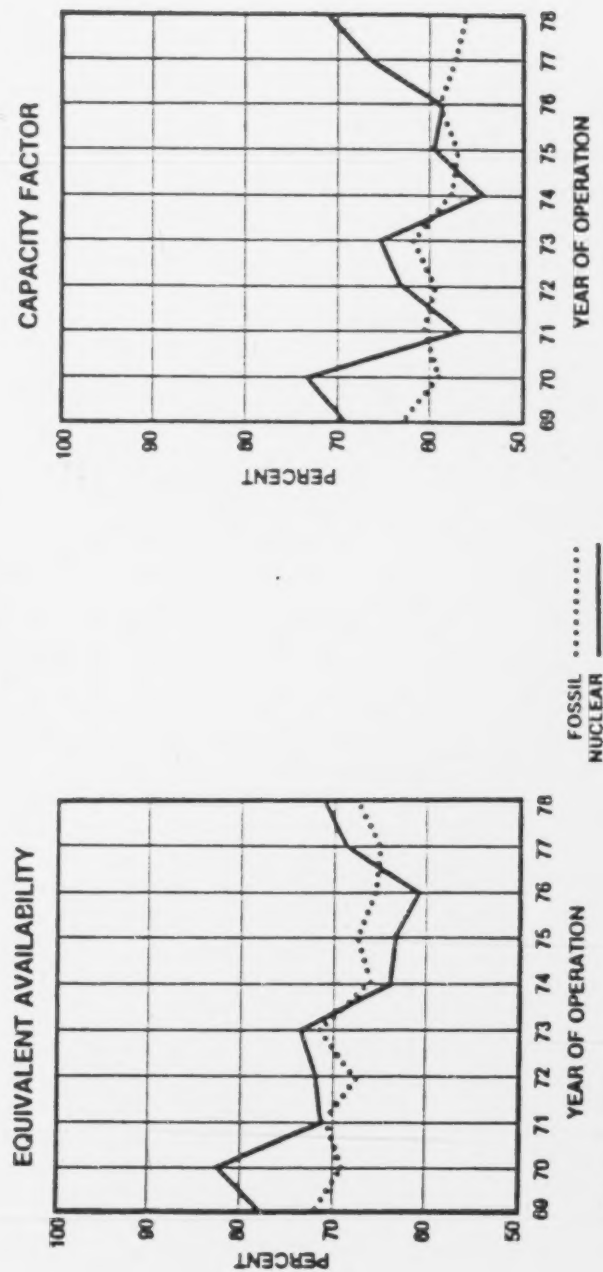
Another widely used basis of comparison is the capacity or load factor**. For the same period, as also seen from Figure 1, nuclear power shows an approximate 4% greater capacity factor. This is a direct result of the substantially lower fuel cost of nuclear power, approximately 18% of generation costs versus 56% for coal, nationally averaged. Where given the choice, a utility will run its nuclear plants before its coal plants.

A somewhat more illuminating comparison made by the Commonwealth Edison Company⁽²⁾ is presented in Table 1.

* Availability factor is the ratio, in percent, of the time a unit is available for service (whether or not it is actually in service) divided by the period time (usually 1 year).

** Capacity factor is the ratio, in percent, of total electric generation divided by the product of the period of time multiplied by the maximum dependable capacity.

UNIT YEAR AVERAGES 1969-1978



Capacity Factor	69	70	71	72	73	74	75	76	77	78	Avg
Fossil	63.1	59.2	60.2	50.7	62.4	57.5	57.2	59.3	57.4	56.7	59.1
Nuclear	69.9	73.3	56.9	63.1	65.1	53.6	59.4	59.0	65.9	70.3	63.4

Equivalent Avail	69	70	71	72	73	74	75	76	77	78	Avg
Fossil	71.9	69.2	71.1	67.2	71.9	66.9	67.2	65.8	65.0	67.3	68.2
Nuclear	78.5	82.5	71.1	71.5	72.8	63.7	63.1	60.6	68.1	70.5	70.2

Figure 1. Comparison of Fossil Coal-Fired* Units 400 MW and Above to All Nuclear Units

TABLE 1

PERFORMANCE HISTORY OF LARGE
COMMONWEALTH EDISON COMPANY COAL AND NUCLEAR PLANTS

	Year								
	1974	1975	1976	1977	1978	1979	1980	1981	AVG.
Availability (%)									
Six large coal	73.0	67.5	59.0	66.7	72.0	64.0	69.8	65.8	67.2
Six large nuclear	65.8	64.2	73.2	79.9	83.6	74.9	73.8	76.5	74.0
Gross Capacity Factor (%)									
Six large coal	55.9	51.8	44.3	47.5	47.2	44.5	47.1	46.3	48.0
Six large nuclear	50.9	50.7	59.6	62.6	70.6	60.5	60.9	64.7	60.0

Commonwealth Edison generates about 45% of its electricity with nuclear plants, one of the largest percentages in the country. On the average, from 1974 to 1981, their nuclear plant capacity factors exceeded coal's by 12%, while the nuclear plant availability exceeded coal's by 7%. Thus on a national average and for a major utility with roughly equal contributions from nuclear and coal-fired power plants the nuclear units outperform the coal units.

It should be noted at this point that there are innumerable tabulations and comparisons of power plant performance, both domestically and world wide, and that different computations of the same quantity, based on essentially the same data, often differ by several percent or more. Therefore, for this study, little reliance is placed on small variations in performance. However, by any measure of significance, nuclear power plants operate at least as reliably as their larger coal-fired counterparts, if not a little more reliably on the average. Moreover, this trend is likely to continue, particularly if the increased use of scrubbers continues to impede the performance of high-sulfur coal units.

The most important point of comparison, however, of nuclear power plants is not their performance relative to coal-fired plants. Rather, it is the consistent variability in performance that is seen between the average units, and those with superior records of performance. To illustrate this consistent variability, using data from the Nuclear Engineering International's tabulation of nuclear plant performance through September 1981 (published December 1981), comparisons were made of cumulative load factors (capacity factors) of a number of reactors and groupings of reactors. The cumulative load factor was chosen as best representing the reactors' long-term performance. Many light-water reactors (LWR's) are

refueling on an 18 month cycle, resulting in annual load factors that are very high (years without refueling) alternating with much lower factors (refueling plus maintenance).

As a benchmark for later comparisons, Table 2 lists the average and weighted cumulative load factors by reactor system type: pressurized heavy-water reactors (HWR's)*, pressurized water reactors (PWR's), and boiling-water reactors (BWR's). It can be seen that, on the average, the performance of the HWR's (with online refueling) substantially exceeds that of both the BWR's and PWR's. It can also be seen that for both the HWR's and the BWR's the weighted cumulative load factor exceeds the average by about 7%, showing that for these plant types the larger units tend to perform better than the smaller ones.

The most interesting information, however, is obtained when examining the performance of individual plants and groups of plants that consistently perform much better than the average. The plant with the highest cumulative load factor is Bruce 3, a Canadian HWR with 83.1% over a 4 year period. Surprisingly, the third reactor is Beznau 2, a Swiss PWR, with 82.9% over a 9 year period. Although five additional non-LWR plants have cumulative load factors greater than 80%, there are 8 LWR's between 75% and 80%, distributed over several countries and reactor vendors. Table 3 lists those U.S. plants with cumulative load factors greater than 65%, or 10 to 20% better than the average.

Two conclusions may be drawn. First, superior performance is not uncommon and second, it is not a matter of chance. The operators of the plants listed in Table 3 are all experienced and most of them operate several nuclear plants. Point

*These are predominantly the Canadian CANDU reactors.

TABLE 2
VARIABILITY OF REACTOR LOAD FACTORS

System	Cumulative Load Factor (%)		Number of Reactors
	Average	Weighted	
HWR	66.7	73.6	13
PWR	56.4	54.5	98
BWR	53.2	60.8	57

TABLE 3
U. S. PLANTS WITH CUMULATIVE LOAD FACTORS
GREATER THAN 65 PERCENT

<u>Plant</u>	<u>Cumulative Load Factor</u>	<u>Type</u>	<u>Years of Operation</u>
Point Beach 2	78.2	PWR	9
Connecticut Yankee	75.9	PWR	13
Prairie Island 2	74.1	PWR	7
Calvert Cliffs 2	73.1	PWR	4
Kewaunee	72.6	PWR	7
Point Beach 1	71.8	PWR	11
Monticello	69.9	BWR	10
Prairie Island 1	69.6	PWR	8
Browns Ferry 3	67.9	BWR	4
Vermont Yankee	67.8	BWR	9
Yankee Rowe	67.8	PWR	20
St. Lucie 1	67.6	PWR	5
Ginna	65.0	PWR	11
San Onofre	65.0	PWR	13

Beach 1 and 2 are operated by Wisconsin Electric Power Company; Prairie Island 1 and 2 and Monticello are operated by Northern States Power Company and the Yankee plants share a common operating experience. A similar example overseas can be seen in Switzerland, where their three PWRs and one BWR average 74.7%, including the extraordinary Beznau 2. One major thesis of this study is that the reliability of the reactor hardware is determined more by the management capability of the operating utility than by the hardware itself. This thesis is supported by several references that describe the management methods used at the Beznau plants. One⁽¹²⁾ in particular, refers to "a well-motivated, trained and qualified plant staff; and a dynamic plant management." A companion report on "Reactor Quality" addresses the complex subject of utility management of nuclear projects.

In order to judge whether or not the past performance of reactor systems is a sound basis upon which to predict their future performance, it is necessary to understand the hardware problems that have been encountered and the ways in which they were addressed by the nuclear industry. A summary of major contributors to past reactor unavailability is presented in Appendix A. Viewing the overall perspective of the past operating history is perhaps more instructive than studying its details. Generally speaking, most, if not all of the major reactor systems and components have experienced some problems that contributed significantly to reactor unavailability. Although this was not anticipated, it should not be surprising, considering the complexity of the nuclear plants, and the generalization that all operational systems, as they start up, undergo a debugging or learning phase. The question that should be addressed regarding nuclear power reactor hardware is: Have the hardware systems matured at an acceptable rate, and is this trend likely to continue?

Examination of the specific problems identified in Appendix A shows them to have been adequately addressed, for the most part, with their contributions to unavailability being decreased or eliminated entirely. A cumulative measure of this can be obtained from the results of two separate studies.

One major finding of an internal DOE⁽³⁾ study in 1980 was that the capacity factors for a reactor's mature years (defined by excluding the first three years of operation) substantially exceeded the lifetime capacity factors. These results are shown in Table 4. The trend is seen to be the same for all U.S. NSSS vendors, with an average difference of over five percent.

A similar study⁽⁴⁾ in 1981 evaluated historical trends in BWR-3 and BWR-4 plants. The principal results, shown in Table 5, are consistent with those of the DOE study. Studies that evaluate the performance of individual reactor types over their lifetimes are more meaningful than the usual tabulations of capacity factors for all reactors, whether tabulated by type or not. The reason for this is that, in the latter tabulation, many newer plants with their immature performance are being continuously added to the data base, thus masking the improved performance of the maturing plants. Hence, the overall impression is given that total reactor performance is improving little, if any. The superposition of non-recurring events, such as the Browns Ferry fire, TMI, or the Stone and Webster seismic analysis problem, further serves to reduce the utility of composite lifetime performance data to identify trends. However, the use of the maturation concept appears to offer consistent evidence that reactor hardware reliability is steadily improving. This is consistent with the impression gained from the review of specific hardware problems; that most problems are effectively remedied, one way or another.

The subject of future nuclear plant availability has been addressed in a systematic way by the reactor vendors. General

TABLE 4

COMPARISON OF LIFETIME AND MATURE YEARS
CAPACITY FACTORS FOR U.S. REACTORS

Reactor Type	Capacity Factors	
	Lifetime	Mature*
PWR	61.0	65.9
BWR	58.7	65.1
NSSS Supplier		
Westinghouse	62.7	68.1
Combustion Engineering	59.7	61.7
Babcock and Wilcox	57.0	61.3
General Electric Co.	58.7	65.1
All Plants	60.2	65.6

*From fourth year of operation

TABLE 5
HISTORICAL BWR-3 AND BWR-4 CAPACITY FACTORS

	Year of Commercial Operation										Composite Capacity Factors	
											Immature Years	Mature Years
	1	2	3	4	5	6	7	8	9	10	1-4	5-10
Unit Capacity Factors	57.3	48.7	56.2	55.2	63.3	67.2	71.0	67.6	65.1	67.4	54.4	66.9
Load-Following and Losses for Economic Reasons	0.0	0.5	0.2	0.6	3.1	1.8	2.9	2.3	7.5	0.6	0.5	3.0
Early Fuel Failure Losses	1.3	0.5	1.3	1.7	1.4	0.0	0.0	1.2	0.0	0.0	1.2	0.4
LPRM Problem Losses	4.4	2.7	0.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.0	0.0
Adjusted Capacity Factors	63.5	52.4	58.4	57.5	67.8	69.0	73.9	71.1	72.1	68.0	58.1	70.3
Unit-Years of Experience	20.0	19.3	19.0	18.6	17.0	14.3	9.3	7.6	3.6	1.2		

Electric Company, in a 1981 paper⁽⁵⁾, discusses the basis for their projection of BWR-6 capacity factors of 80%. They have identified specific design improvements that should add 12% to the 1979 U.S. BWR average capacity factor of 68%. Similarly, although not in such detail, Westinghouse Electric Corporation⁽⁶⁾ describes its plans for an advanced PWR, with a plant availability goal of 90%. Both papers emphasize similar efforts to improve hardware reliability through improved designs and corrosion-resistant materials, changes to allow improved maintenance at lower radiation exposure levels, and extended periods between refuelings. Based on the continuation of present trends, their expectations are not unreasonable. In addition, there is increased emphasis on comprehensive design reviews of hardware designs followed by extensive testing of hardware before its commercial use. Except for a few very large components with very long service life, adequate testing should identify problems before they affect plant reliability. It is interesting that NASA's approach to reliability⁽⁷⁾ is now roughly the same, compared with its early emphasis on Failure Modes Analysis and detailed component reliability studies.

However, as will be discussed later, there are still several unresolved hardware problems that must be considered, as well as the possibility that new, unexpected problems will arise. Also, the performance of the BWR-6 units and CE System 80 units just coming on line must be closely monitored to see if new problems were introduced as the old ones were being fixed.

The Cost of Maturation

Even if it is granted that nuclear plants are maturing, it is valid to ask, "At what price?" Determining the costs required to fix the past reactor hardware problems is a difficult task.

Except for a few very large problems, such as steam generator replacement at approximately \$60 to \$80 million, or the Browns Ferry 1 and 2 recabling, most major repairs, replacements or modifications cost less than \$10 million--with the vast majority costing much less still. These costs are distributed over vendors, various users groups, EPRI, and individual utilities. Also the costs of replacement power are not included in the cost of repairs, which, for the utilities, usually shows up as Operating and Maintenance (O&M) costs--except for the larger items, which are usually capitalized. Also, accounting procedures used by utilities vary, as do the PUC rules under which they operate--making O&M costs an often inconsistent measure. Nevertheless, the best way to gauge the costs of maintaining nuclear power plants is to compare them to their closest analogy, the larger base-loaded coal-fired plants. In fact, there have been innumerable studies, comparing past O&M costs of coal and nuclear power, and predicting their future trends. The results of one 1981 summary⁽⁸⁾ of several such studies is presented in Table 6. It must be noted that the various studies are usually compiled on different bases, but the general consensus appears to be that future O&M costs for coal plants will likely exceed those of nuclear plants. This trend can also be seen from results quoted in the previously referenced Commonwealth Edison Company paper⁽¹⁾, which were for large units operated by the same utility. The trends shown in Table 7 indicate that as of today, coal O&M costs are increasing at a faster rate than nuclear O&M costs. However, in comparing O&M costs, one must be careful to consider the interplay between costs expressed as mills/kwhr, and capacity factor, which indicates how much power was produced. When a plant is down through planned or forced outages, its O&M costs in actual dollars go up because of the added activity to refuel or to perform maintenance or to remedy problems, and its power output goes down because of the outage, thus doubling the impact on mills/kwhr. Two other factors should also be considered before placing too much

Table 6
Estimated O&M Costs for Large Central Station Power
Plants, Nuclear and Coal (with scrubbers)

	NASAP (1)	Source of Estimate			Sargent & Lundy (6)	Stone & Webster (7)	Charles Kormanoff		
		EBASCO (2)	UE&C/DBRL (3,4)	Gibbs Hill (5)			Power Plant Cost Escalation (1981) (8)	Comments (9)	Testimony (10)
O&M Cost for Nuclear, m/Kwh	2.2	7.5	1.9 3.8	3.2	19.2	5	6.2	46.2	57.1
O&M Cost for Coal, m/Kwh	3.0	8.6	2.9 3.9	4.3	27.1-33.4	6	6.2	27.7	33.1
Assumed Capacity Factor, %	70	65	70 70	60	60	60	70(c) 60(n)	70(c) 60(n)	70(c) 60(n)
Year of Commercial Operation									
Nuclear	1978	1992	1978 1981	1979	1992	1990	1988	1988	1988
Coal	1978	1990, 1992*	1978 1981	1979	1992	1990	1988	1988	1988
Cost Basis	Constant 1978 \$	10 yrs. levelized current \$	con- stant, 1978 \$ 1981 \$	10 yrs. levelized 1979 \$	30 yrs. levelized current \$	5 yrs. levelized 1990 \$	30 yrs. levelized 1979 \$	30 yrs. levelized current \$	year 2000 center of gravity, current \$
Coal/Nuclear Ratio	1.36	1.15	1.53 1.03	1.34	1.41-1.24	1.20	1.00	0.60	0.58

* Start-up dates for dual units

(c) Coal

(n) Nuclear

- (1) DOE/NE-001/5, "Nuclear Proliferation and Civilian Nuclear Power ", Vol. V, June 1980.
- (2) D. J. Kettler, "Economics of Future Generation: Nuclear Versus Coal Post-TMI", Transactions of the ANS, Vol. 38, June 1981.
- (3) COO-4954-1, "Final Report and Initial Update of the Energy Economic Data Base (EEDB) Program, Phase I", Dec. 1979.
- (4) COO-6411-1, "Phase IV Final Report and Fourth Update of the Energy Economic Data Base (EEDB) Program", Sept. 1981.
- (5) "Economic Comparison of Coal and Nuclear Electric Power Generation", Gibbs & Hill, Inc., Jan. 1980.
- (6) W. W. Brandtson, "Cost Comparison of Central Electric Power Generation Using Coal and Nuclear Fuels", (CONF-800403), Vol. 2, April 1980.
- (7) W. J. L. Kennedy and R. W. Campbell, "The Economics of Nuclear Power", for the Atomic Industrial Forum Conference on Nuclear Power and the Public, Kansas City, MO, Feb. 26, 1979.
- (8) Charles Komanoff, "Power Plant Cost Escalation, Nuclear and Coal Capital Costs, Regulation and Economics" at 267 (Komanoff Energy Associates 1981).
- (9) Comments of C. Komanoff concerning the Theodore Barry Associates Audit of Nine Mile Point 2 Project Cost Estimate and Construction Plans, submitted to the New York Public Service Commission, August 25, 1981.
- (10) Direct Testimony of Charles Komanoff In PSC 28059 (submitted Nov. 6, 1981).

TABLE 7

COMPARISON OF O&M COSTS FOR LARGE
COMMONWEALTH EDISON COAL AND NUCLEAR PLANTS

	Nuclear		Coal	
	<u>6 largest units</u>		<u>6 largest units</u>	
	<u>1976</u>	<u>1981</u>	<u>1976</u>	<u>1981</u>
O&M Costs (mills/kwhr)	2.5	4.2	1.7	5.0
Capacity Factor (%)	59.6	64.7	44.3	46.3
O&M Costs Corrected To 60% Capacity Factor	2.5	4.5	1.3	3.9

emphasis on extrapolation of past O&M trends. First, increasing use of scrubbers to reduce sulfur in flue gas is likely to substantially increase coal O&M costs, based on the limited operating experience to date. Second, nuclear O&M costs include factors unrelated to the reliability of plant hardware, such as site security and regulatory-related analytical support. We believe that costs relating to these activities will not increase indefinitely.

However, based on the continuation of present trends, it is unlikely that the reliability of nuclear plant hardware will be a negative economic factor in future decisions to choose a system for the base-load generation of electricity. Measured by both the plant availability and the cost to maintain the hardware, nuclear plants are likely to perform at least as well as, if not measurably better than, coal-fired alternatives.

Potential Hardware Reliability Issues

As mentioned previously and discussed in more detail in Section 3.0, a number of major hardware-related issues are identified that could adversely affect the future performance of nuclear reactors. These are Pressurized Thermal Shock, Fires, Seismic Events, Control Systems, Plant Safeguards, Systems Interaction, Hydrogen Control and Effects of a Hydrogen Burn, Shutdown Decay Heat Removal Requirements and Internal Flooding. These issues may have significant impacts through three mechanisms: 1) major plant changes as a result of NRC-mandated requirements, 2) utility-identified changes to reduce their financial risk and 3) a major incident and an after-the-fact response by both. The problem facing the industry is to avoid 3), by accomplishing 1) and/or 2) in a cost-effective manner, if they are indeed necessary.

Using the issue of pressurized thermal shock (PTS) as an example, it is seen that: 1) it is not clear that there is a problem, 2) if there is a problem, there are several methods of resolution having various cost impacts, and 3) the problem only affects a limited number of reactors. This generalization holds true for most of the potential problem areas discussed in Section 3.0.

For PTS, potential solutions include:

- o In-situ annealing of the pressure vessel
- o Derated plant operation
- o Increased temperature of ECCS water to reduce thermal shock
- o Improved control and instrumentation systems and improved operator procedures to reduce likelihood
- o Modified fuel loads to reduce neutron fluence at the vessel welds

More importantly, there are significant uncertainties in the analytical methods used to calculate the overcooling transients and their effects on the vessels in question. Also there are $\pm 50\%$ uncertainties in fast-neutron fluence data.

A reasoned assessment of the overall PTS issue, allowing for improving the state of knowledge over the next few years, leads to the conclusion that it will not have a major impact on the future of nuclear power. It is likely to have, at the worst, a limited effect on a few older plants and none at all on future plants and those of recent vintage. However, even

if all presently identified issues of concern were resolved in a similar manner, with acceptable impact on future plant performance, there would still remain the nagging doubt that a new major end-of-life problem will surface.

A review of past reactor history should give some assurance that it is unlikely. Present LWR systems are evolutions of commercial systems that have accumulated about 2000 reactor years of operation, with over 160 reactors now in service worldwide. The oldest have operated for over 20 years, and in several instances, the older plants are some of the most reliable. Problems have been experienced and resolved in most reactor systems. Also most reactor components are designed for some maintenance during their service life. It is only the very large passive components (e.g., vessels, internals, steam generators, and structures) and the piping and wiring that were intended to last the full life of the plant⁽⁹⁾. Nevertheless, the industry has been able to accommodate, at some financial pain, repairs to these. There is no reason to believe that future plants will be less reliable than present ones, or that future problems will be worse than past problems.

Conclusion

The likely future of nuclear plant reliability is that the hardware will not be trouble-free, but will continue asymptotically on the trend of improved performance, towards a practical maximum, typical of all complex large scale facilities. The continued close surveillance and inspection are likely to identify problems before they have serious impacts on plant performance or the public's health and safety. The rate and extent of improvement of hardware reliability will depend on the industry's ability and willingness to learn from past

mistakes and to emulate the operating practices of those plants exhibiting superior performance.

With the continuation of these trends, it should be noted that any new power-producing technology, nuclear or otherwise, will have to compete against the hardware performance of evolutionary LWR's as it will be, not as it was. Also, all power plants will share the common problems associated with their secondary sides--the turbine-generators and the condensers and feedwater and condensate systems, for those plants using steam cycles. It is difficult to see how the reliability of hardware performance will become a major issue affecting the future of nuclear power, if it is viewed objectively by the decisionmakers and not overly complicated by regulatory intervention.

The study of historical reactor hardware reliability discussed previously was extended to present and future reliability concerns. This effort consisted of a review of published documents, such as, ACRS and NRC reviews of generic and unresolved safety issues, the results of probabilistic risk assessment studies which identify important contributors to nuclear plant risk, reports regarding accident precursors, evaluations of Licensee Event Reports (LERs), and personal communications with technical experts studying such problems. In this manner, current or potential problems, which might emerge as technical issues in future years, and that could become serious enough to cause difficult problems for the nuclear industry were identified. This effort resulted in the identification of nine such problems which are currently being analyzed by the industry, but which have not yet been resolved. These are: Pressurized Thermal Shock, Fires, Seismic Events, Control Systems, Plant Safeguards, Systems Interaction, Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment, and Shutdown Decay Heat Removal Requirements and Internal Flooding. The following sections contain brief summaries of each of these unresolved issues, including the possible implications for the nuclear industry.

Pressurized Thermal Shock

Pressurized Thermal Shock (PTS) is one name for a hypothetical reactor accident in which a severe reactor system overcooling event is followed by repressurization of a reactor vessel whose fracture toughness has been reduced due to neutron fluence, thus causing extensive propagation of crack-like defects in the vessel wall, compromising reactor vessel integrity. The reactor-system overcooling events can result from

a variety of causes that include instrumentation and control system malfunctions and postulated accidents such as small break loss-of-coolant accidents (LOCA's), main steam line breaks (MSLB's), feedwater pipe breaks, or stuck-open valves in either the primary or secondary system. Rapid cooling of the reactor vessel internal surface causes an increased temperature gradient across the reactor vessel wall. The temperature distribution results in thermal stress, with a maximum tensile stress at the inside surface of the vessel and a compressive stress at the outside surface and the magnitude of the thermal stress dependent on the temperature differences across the reactor vessel wall. These stresses combine with the hoop stress caused by the internal pressure in the vessel.

As long as the fracture resistance of the reactor vessel material remains relatively high, such events will not cause failure. However, after the fracture toughness of the vessel is reduced by neutron irradiation, several thermal transients could cause fairly small flaws to propagate near the inner surface and result in significant, and possibly through-the-wall, cracking. The vessels of concern are those which have accumulated high neutron radiation exposure and which are made of material that has a high sensitivity to neutron irradiation (such as those made with welds of high copper content).

In order to threaten the adequacy of core cooling by pressurized thermal shock events, a number of contributing factors must be present. These factors are: 1) a reactor vessel flaw of correct size to propagate, 2) high copper content welds, 3) a relatively high level of irradiation, and 4) a resulting crack of such size and location that the ability of the reactor vessel to maintain core cooling is affected.

In view of operational experience and post-TMI attitudes which favor keeping the reactor coolant pumps running to provide forced circulation cooling in case of an accident or trans-

ient, the NRC believes that a PTS program is needed which will formulate a regulatory requirement to ensure that the risk of pressure vessel failure from PTS events is sufficiently low through each vessel's design lifetime. The program that will be conducted to provide firm bases for such a regulatory requirement includes: development of methods for estimating the probability and severity of PTS transients and the operator's role in such events; refinement of methods for determining pressure vessel stresses in the event PTS transients do occur; refinement of methods for determining material properties and failure vulnerability of the vessel due to PTS stresses as a function of vessel exposure to neutron irradiation (and thus as a function of time in plant life); evaluation of potential benefits from possible corrective actions; and development of criteria for acceptability of plant safety margins under postulated PTS events. This program will provide information to aid in assessing acceptability of several PTS studies currently underway in the industry, as well as forming a basis for recommending acceptance criteria for resolution of the PTS issue.

Among the various corrective actions which have been proposed to date to resolve the PTS problem are the following:

- (1) Modified fuel loading patterns which would reduce fast neutron flux at the reactor vessel wall.
- (2) Increased temperature of emergency core cooling system injection water.
- (3) Improved control and instrumentation systems.
- (4) Improve procedures and training to facilitate appropriate operator actions.

(5) Reactor vessel annealing.

(6) Derated plant operation.

Of the potential corrective actions identified, numbers (3), (5), and (6) represent possible serious adverse impacts on unit availability. In particular, (5) could result in capacity factor losses and costs that far exceed any corrective action yet taken by the industry, including the replacement of steam generators.

Fires

This issue deals with the threat that control system and cable fires pose to the industry. Possible consequences of fires to utility financial investment may be far greater than any threat to public health and safety. As an example, suppose a sizeable cable fire were to occur at a plant taking out one entire train of a redundant set of cables, thereby not seriously compromising the safety functions, but causing the plant to shut down for a year or more to replace the affected cables. Such an event would cause significant capacity factor losses because of the resulting lengthy outage and would likely be very costly, in terms of replacement materials and labor.

Seismic Events

The recent findings from the Zion and Indian Point probabilistic risk assessment (PRA) studies that seismic events are dominant contributors to risk may or may not be accurate. For the purpose of this discussion, it does not matter. In a scenario in which a rather modest earthquake (0.10g ground acceleration) strikes a reactor in the near future, such an event would probably cause only obvious superficial damage to the plant, with no real threat to public health and safety.

However, such an incident would no doubt result in an extended shutdown (many months), while every system and component in the plant is reanalyzed and reinspected before the plant is allowed to resume operation. Again, this would result in significant capacity factor losses due to the extended outage, and fairly high costs, because seismic analyses such as that performed on the Diablo Canyon plant are expensive and take many months to perform.

Control Systems

This issue deals with the concern that the failures or malfunctions in the control systems have the potential of causing postulated accidents or transients to become more severe than analyzed. The failures or malfunctions may occur independently or as a result of the postulated accident or transient. One aspect of the concern is the potential for a single failure, such as loss of power or short circuit to cause simultaneous malfunctions of several control features. The other aspect of the concern is for a postulated accident to cause control system failures or malfunction which would make the accident more severe than analyzed. For example, accidents could cause control system failures either by creating a harsh environment in the area of the control equipment or by physically damaging the equipment.

The effects of control system failures are broadly categorized as follows:

- o The effects of control system failures on anticipated operational occurrences.
- o The effects of control system failures on accidents.
- o The effects of control system failures on operator actions.

- o The effects on minor radioactive leaks.

It is generally believed that the failures or malfunctions in control systems are not likely to result in loss of safety functions that could lead to serious events or result in conditions that could not be handled by the safety systems. However, in-depth studies have not been performed to support this belief.

The NRC has established a generic program to resolve this issue. Its overall objectives are:

- o To assess the safety implications of control systems and attendant plant characteristics.
- o To formulate a methodology for assessing the modes and effects of failures of control systems on the basis of common cause, common mode, and other multiple failures, such as cascade failures.
- o To develop criteria for establishing the relative importance of control systems important to safety and recommend design and operational criteria for these systems based on their importance to safety.

One of the major subtasks of the program defined by the NRC is to study the control system failures leading to steam generator overfill transients in pressurized water reactors and reactor overfill transients in boiling water reactors. The second major subtask that has been identified is the evaluation of control system failures leading to reactor overcooling transients. The third subtask is the evaluation of all other control system actions that have safety implications. The

fourth subtask is planned for studying the effects of the loss of power to control systems. It is expected that the studies will lead to the determination and definition of the need for preventive and mitigative design measures to accommodate the transients.

The subtasks will address measures to improve the control systems reliability, to reduce the effects of their failures and to improve the capability of coping with effects of the failures.

It is anticipated that following the final resolution of the issue systematic evaluation of control system designs may have to be performed to evaluate safety implications on a plant-specific basis. It is too early to predict the extent to which reviews may be required and the impact of such reviews.

In parallel with the study on control systems, the NRC staff plans to evaluate the plant electrical systems that provide power to the control systems. The overall objectives of the electrical system evaluation program are:

- o To develop a technique for assessing the performance of electrical systems, including their power sources, controls and distribution, when they are exposed to component failures, operator-induced failures, and to abnormal energy sources and sinks.
- o To develop a technique for assessing the effect of modifications to the electrical systems required by the NRC or by others.

It may be too early to speculate how great an impact the resolution of this issue could have on the industry. However, it is possible that the investigation which the NRC has undertaken might discover some major problems with control systems, especially those that are more "automated", when they are viewed in the light of "cognitive human error" problems. If, as a result of the NRC study, design changes to plants are required, such changes could cause significant capacity losses. The extent of any losses would depend on the length and timing of the changes, but it is possible that some or all of the changes could be effected during a refueling outage. Whether this approach would be acceptable would depend on the identified public risk due to the potential control system problems.

Plant Safeguards

In spite of the emphasis that has been placed on this issue since the NRC reported it to Congress in 1978 as a Category A item, the possibility still exists that an individual or individuals will threaten a nuclear plant, either verbally or physically. Such a threat might even lead to physical harm to a plant, although such actions might not pose a threat to public safety. This event could result in an extensive outage, depending on the extent of the physical damage. The capacity factor losses which might result would again be a function of the length of time the plant was out of service, with the costs being proportional to the length of the outage and the extent of any required repairs.

Systems Interaction

The NRC review and evaluation of plant systems is performed in accordance with the Standard Review Plan which organizes the task either on a system-by-system basis or by engineering discipline. Each element of the Standard Review Plan is assigned

to an organizational unit for primary responsibility and, where appropriate, to other units for secondary responsibility. Thus, the review process does take into consideration interdisciplinary areas of concern and accounts for systems interaction. Furthermore, many of the regulatory criteria are aimed at controlling the risks from systems interactions. Examples of such criteria include the single failure criterion and separation criteria.

The design of the nuclear power plant is also accomplished either on a system-by-system basis or by engineering disciplines. The reviews performed on the design include interdisciplinary considerations, such as the compatibility of structures, components, and systems to ensure that the functional requirements will be satisfied at system level. Safety and accident analyses further ensure that the functional requirements of the system are met. Such analyses include failure mode and effects analyses. Therefore, the design and review process does take into consideration systems interactions to a large extent.

In spite of the existing mechanisms in the design and review process, concerns remain regarding the interaction of various plant systems, both as to the supporting roles these systems play and to the effect one system can have on other systems, particularly with regard to whether actions or consequences could compromise the engineered redundancy and independence of safety systems. The adverse interactions could inhibit the ability of safety systems to function either for mitigating the consequences of an accident or for achieving a safe shutdown.

The objectives in resolving the issue are, to identify where the present design, analysis, and review procedures may not

adequately account for potentially adverse systems interactions and to recommend the regulatory action that should be taken to rectify deficiencies in the procedures.

The implications of the resolution of this issue are that every utility may be required to perform a plant-specific evaluation of systems interaction. Such an evaluation could result in the need for plant design modifications which would require plant shutdown for implementation. The extent of the capacity factor losses incurred as a result of system interaction problems would depend on the length of any required outages and whether the necessary plant modifications could be performed during a refueling outage. Associated costs would be directly proportional to the length of the outage and to the labor and equipment required to implement any necessary modifications.

Hydrogen Control and Effects of a Hydrogen Burn

Following a LOCA in an LWR plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (1) metal-water reaction involving the fuel element cladding; (2) radiolytic decomposition of the water in the reactor core and the containment sump; (3) corrosion of certain construction materials by the spray solution; and (4) synergistic chemical, thermal, and radiolytic effects of postaccident environmental conditions on containment protective coating systems and electric cable insulation.

In the event of a degraded or melted core, a large additional amount of hydrogen would be generated as a result of a reaction between the molten fuel and the concrete containment base. Other combustible gases may also be generated by this reaction.

Because of the potential for significant hydrogen generation as the result of an accident, Title 10 of the Code of Federal Regulations (10 CFR) Section 50.44, "Standards for Combustible Gas Control Systems in Light Water Cooled Power Reactors," and General Design Criterion 41, "Containment Atmosphere Clean-up," in Appendix A to 10 CFR Part 50 require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained. Conventional hydrogen control systems (for example, hydrogen recombiners) have historically been installed to provide the capability to control the hydrogen accumulation from radiolysis, corrosion of metals inside containment, and environmental effects on coatings and insulation. The design capability or margin to control the contribution to hydrogen accumulation resulting from a metal-water reaction involving the fuel cladding has historically been provided by the net free volume inside the containment structure. That is, the containment volume was large enough such that hydrogen generated and released from the cladding reaction would not reach a uniform concentration approaching the lower limit of flammability. The reason for this approach is that the rate of hydrogen release as a result of a cladding reaction is assumed to be rapid following a postulated accident (that is, on the order of minutes). This corresponds to a release rate well beyond the capability of conventional hydrogen control systems. However, the containment net free volume was found to be sufficient for providing the initial protection, and hydrogen control systems would be actuated later to control hydrogen accumulation from the other sources and gradually reduce the hydrogen concentration inside containment.

10 CFR Section 50.44 requires that the combustible gas control systems provided be capable of handling the hydrogen generated as a result of degradation of the emergency core cooling system (ECCS) in such a way that the hydrogen release is five times the amount calculated in demonstrating compliance with 10 CFR Section 50.46 or the amount corresponding to reaction of the cladding to a depth of 0.00023 inch, whichever amount is greater.

The accident at TMI-2 resulted in metal-water reaction which involved hydrogen generation well in excess of the amounts specified in 10 CFR Section 50.44. As a result, it became apparent to the NRC that additional hydrogen control and mitigation measures would have to be considered for all nuclear power plants.

Pending long term resolution of the hydrogen control issue, the NRC has embarked on an interim rulemaking procedure which has recently published a final rule (46 FR 58484) and a proposed rule (46 FR 62281). The final rule, published December 2, 1981, consisted of the following major provisions:

1. Inerting of Mark I and II BWR's,
2. Internal recombiners or capability to install external recombiners for all LWR's, and
3. High point vents in the reactor coolant system for all LWR's.

The proposed rule, published December 23, 1981, contained the following staff recommendations:

1. BWR Mark III and ice condenser containment plants have H₂ control systems capable of handling a 75%

clad/water reaction (presently a distributed ignition system has been approved as an interim solution at Sequoyah),

2. Capability of safety equipment to survive a hydrogen burn for all LWR's that are not inerted, and
3. LWR's (except inerted plants) perform analyses evaluating large quantities of H_2 (including control system considerations) and containment capabilities.

The resolution of this issue has ominous overtones for the nuclear industry, especially for those plants currently in operation. Some of these plants have no provisions for hydrogen recombiners. If backfitting of such capability is required, these plants may have to undergo extensive modifications with corresponding outage times and capacity factor losses. There are also the possible costs of adding recombiner capability to a containment that currently has no penetrations for such equipment. In addition, under the recent regulatory requirements, a number of new plants may find that their containments are unable to withstand the effects of ignition of a large quantity of hydrogen.

Shutdown Decay Heat Removal Requirements

When reactor shutdown occurs, measures must be available to remove decay heat from the reactor. This is accomplished in LWR's by satisfying two heat removal functional requirements: (1) ensure adequate cooling of the fuel by maintaining a sufficient water inventory in the reactor vessel, and (2) provide a means of transferring decay heat from the reactor coolant system to a heat sink. The reliability of performing these functions depends on the frequency of events that require or

jeopardize decay heat removal operations and the probability that necessary systems will be available to remove the decay heat.

One of the important factors in the safety of nuclear reactors is the reliability of the systems used for decay heat removal following the shutdown of the reactor for any reason. The results of the Reactor Safety Study (WASH-1400) showed that the overall probability of core meltdown in the first generation of large commercial LWR's was higher than had been expected (about 5×10^{-5} as compared to 1×10^{-6} per reactor year). Insufficient reliability in the decay heat removal system, particularly in response to small loss-of-coolant accidents (LOCA's), was shown to be responsible for a substantial portion of the overall probability of core meltdown.

The NRC Staff feels that providing an alternative means of decay heat removal could substantially increase the capability to deal with a broader spectrum of transients and accidents, and therefore, could potentially significantly reduce risk to the public. Consequently, the plan for resolution of this issue will investigate alternative means of decay heat removal in PWR plants, including but not limited to using existing equipment where possible. This study will consist of a generic systems evaluation and will result in recommendations regarding (1) the adequacy of existing shutdown decay heat removal requirements, and (2) the desirability of, and possible design requirements for, an alternative decay heat removal method, that is, a method other than that normally associated with the steam generator and secondary system. The plan for resolution will also investigate the need and possible design requirements for improving reliability of decay heat removal systems in BWR's.

The implications of the resolution of this issue will be quite significant if it is determined that alternative means of

shutdown decay heat removal must be provided for every plant. The capacity factor losses that would be incurred in this case will be a function of the extent of the outage necessary to install the additional means of decay heat removal. If such installation can be effected during a refueling outage, the losses would be minimized. The costs of an alternate system will depend on the complexity of its design and how readily it can be added to the existing plant configuration.

Internal Flooding

In this case, the problem is that most internal flooding analysis is concerned with flooding from piping ruptures or seal failures that lead to a discharge at a specific rate for a specific length of time. This results in a specific quantity of water in the plant which submerges particular safety equipment. As with the case of fires, the defense-in-depth design philosophy probably affords adequate protection to public health and safety. However, there may be a much greater risk to utility investment from floods that pose little risk to the public. A vivid example of this problem was the failure of a service water line supplying the containment cooling system at the Indian Point plant. To date, no resolution for this type of flooding situation has been proposed. Periodic inspections and early detection capability offer the only hope that any such flooding, if it occurs, will be held to a minimum.

MECHANISMS FOR IDENTIFYING POTENTIALLY SIGNIFICANT HARDWARE ISSUES

Potentially significant issues arise from a variety of information sources, and the method chosen to evaluate the information or to resolve the issue may cause the issue to develop into a major technical problem.

One function of the Nuclear Regulatory Commission's Office of Inspection and Enforcement (I&E) is to notify licensees and permit holders regarding matters of safety, safeguards or environmental significance. This is accomplished through the issuance of I&E Bulletins, Circulars and Information Notices, of which only Bulletins require a response in all instances.

Bulletins

When the NRC becomes aware of inadequacies or defects in a facility that seem to be of a generic nature, I&E may issue a Bulletin to a group of licensees requiring inspection, reporting, and correction or commitment to correct the situation as appropriate. The Bulletin will specify the action required of each class of licensee, and the schedule for completion of the action. Also, the Bulletin will specify reporting requirements (in some cases, reporting is not required and inspectors will review the response at the individual facility).

The I&E Manual provides for an order modifying the license of any licensee not responding to the action requirements of a Bulletin. Thus, an I&E Bulletin carries legal force.

Circulars

Occasionally the NRC is informed about a situation which does not meet the criteria for a Bulletin either in significance or immediacy, but could have potential generic implications. In

these cases an I&E Circular is issued. Generally, a Circular recommends that licensees review the information provided and take certain specific preventive or review actions. Since the Circular identifies potential problems not of immediate health and safety significance, no reporting is required. Thus, the Circular is basically a mechanism for the dissemination of information on problem areas and recommended licensee actions.

Information Notices

The newest category of Inspection and Enforcement communication is the Information Notice. The I&E Information Notice provides "early notification of a possible significant matter." It is issued when the NRC becomes aware of a potentially significant matter but does not yet know the extent or seriousness of the deficiency. Licensees are expected to review the information for applicability to their facility, but no specific action is required and no response is required. If further evaluation warrants additional licensee action, the matter will be upgraded by issuance of a Circular or Bulletin.

Although the Circular and Information Notice require no formal response, the Office of Inspection and Enforcement has indicated that licensee review of these documents and resultant actions are subject to inspector audit.

Generic Letters

When an issue is raised which affects a specific type or group of plants, the NRC may issue a generic letter to those involved. The NRC began issuing generic letters in 1977 as a means of reducing the amount of paperwork necessary to disseminate information to a large number of plants. In January 1981, a numbering system was devised which facilitates easier tracking of a specific letter. The generic letters may

request information, transmit information or forward documents.

Licensee Event Report (LER)

Certain reportable events as described in the Technical Specifications are required to be reported to the NRC. These are transmitted to NRC as a minimum on the LER form to provide a complete explanation of the circumstances surrounding the event. From these reports equipment reliability trends can be developed for predictions of future problems.

Industry Resources

The nuclear industry has a number of organizations and services that gather and review information and that analyze and disseminate varying amounts of operations and maintenance data. Among these are the Nuclear Operations Maintenance Information Service, the Nuclear Safety Analysis Center, the National Electric Reliability Council, just to name a few. These services distribute information on hardware performance via reports, meetings, computer listings and statistical data summaries.

Nuclear Operations Maintenance Information Service, (NOMIS) is a forum open only to utilities. Its purpose is to increase nuclear power plant availability and maintenance efficiency through the timely exchange of operations and maintenance information at a utility working level.

Nuclear Safety Analysis Center, (NSAC) is operated for the electric utilities industry by the Electric Power Research Institute. NSAC provides ongoing review and analysis services of licensee event reports (LER's), and assesses potential generic problems and possible solutions. NOTEPAD was developed by NSAC as a part of its function to serve as a clearinghouse for

technical information on nuclear safety and licensing. NOTE-PAD is a computerized teleconferencing system for the mutual exchange of information. Users are connected to a central computer by portable terminals and telephone couplers.

National Electric Reliability Council (NERC) presents statistical data on the performance of major types of electric power generating units through the Generating Availability Data System (GADS). While focusing on improving the accuracy, completeness and timeliness of reporting, the GADS program also stresses practical data applications in solving problems encountered by electrical and mechanical power plant operators and designers.

In addition, the Institute of Nuclear Power Operations (INPO) now performs the management and technical direction of the Nuclear Plant Reliability Data System (NPRDS). NPRDS, originally developed in the early 1970s by the Edison Electric Institute, gathers engineering and failure data on safety-related systems and components. NPRDS is coupled effectively with INPO's Significant Event Evaluation and Information Network (SEE-IN) to provide the nuclear utilities, NSSS suppliers and A/E firms with a board-based, user-oriented feedback system.

In considering the future of nuclear power, one must address the question whether new, serious problems which are not considered important today are likely to emerge. Based on the review of operational experience performed for this study it appears that "new" problems are inevitable, in the same sense that a number of the problems discussed previously were not a cause for concern in the early 1970's. There are two reasons for this observation: (1) reactors are aging, (2) the industry is getting smarter and is obtaining an experience data base that is revealing subtleties.

If the current regulatory environment is allowed to continue its present course unchanged, the aging issue could be detrimental to the future of nuclear power. This means that, although age brings deterioration (true of reactors as well as people), it also brings experience and wisdom (also true of reactors as well as people). The NRC's system has little accommodation for balancing these two effects for reactors. As a result, "safety problems" will be identified with hardware all too easily. Because credit for experience or "wisdom" will probably not be quantifiable, the tendency to "ratchet" new requirements will prevail in the future.

On the positive side, the industry's ability to cope effectively with new problems will improve dramatically with the passing years. The interaction among utilities is much better now -- witness the number of owners' groups that are currently striving to solve a number of licensing issues.

Although it may be possible to determine semi-quantitatively how much "smarter" plant operators have become, there isn't much hope that anything quantitative enough to be really convincing can be done.

As an example of how the industry has become "smarter" with increasing age, consider a recent analysis of BWR availability and capacity factor.⁽⁵⁾ Through this analysis, General Electric Company (GE) shows that much progress has been achieved in fully understanding and correcting problems that had adversely affected BWR unit availability in the past. GE performed this analysis to forecast the availability and capacity factor for its most recent design, the BWR/6. The analysis was performed in the following way. First, the causes of lost availability and capacity factor for U.S. BWR/2-4 operating units during a reference year, 1979, were identified. Then the design differences between the operating BWR/2-4 plants and the BWR/6 were identified. Next, the contribution of each design difference toward changing availability and capacity factor for the BWR/6 relative to current operating BWR/2-4 plant experience was quantified. The BWR/6 availability and capacity factor forecast was then formed by adding the availability and capacity factor changes for the BWR/6 to the availability and capacity factor for operating BWRs in 1979.

The performance for new BWR/6 during the first few years of operation was also estimated in the GE analysis and compared with corresponding data for BWRs operating in 1979. The results indicated that, for the first few years of operation, BWR/6 availability and capacity factor performance are expected to be lower than those of a mature BWR/6 with several years of operating experience. However, new BWR/6 plants were projected to be capable of higher availabilities and capacity factors than the plants operating in 1979 had during their first few years of operation. As a result, mature BWR/6 plants were projected to have capacity factors averaging 80% and availabilities that averaged 85%.

The specific design improvements for the BWR/6 relative to BWR/2-4's operating in 1979, and the associated capacity fac-

tor change forecast for each category are shown in Table 8. Table 9 shows the projected improvement in capacity factor for the BWR/6 over the capacity factor reported for BWR/2-4 during 1979. As can be seen from Table 9, the BWR/6 capacity factor is projected to be about 80%, a 12% improvement over the BWR/2-4 performance for 1979.

No analogous quantitative analysis by any of the three PWR vendors was available, but Westinghouse Electric Corporation in a recent paper⁽⁶⁾ described their development program for an advanced PWR with a 90% availability. Thus it appears that the industry believes that present trends in improved hardware reliability can be continued.

From the information reviewed for this study of the future of conventional power the improved reliability performance is a very reasonable expectation. While unforeseen problems are inevitable, there is no evidence to indicate that they cannot be resolved in an acceptable manner, provided decisions are made in a rational licensing framework.

A considerable investment in both time and money is required to complete either a nuclear or a fossil-fueled power plant, once the decision to build the plant has been made. The utility expects to realize a reasonable financial return from the investment in the plant. In addition, the utility has certain expectations for the power-producing performance of the plant and the security of the investment represented by the plant. There is currently increasing interest and activity within the utility industry to improve fossil and nuclear plant productivity. Included in these efforts are applications of the RAM (Reliability, Availability, and Maintainability) methodology. Although there is a great deal of interest, very little methodology has been developed in the area of risk of investment

TABLE 8
BWR/6 DESIGN RESPONSE TO OPERATING EXPERIENCE*

Operating Plants	BWR/6	
Source of Capacity Factor of Loss	Design Improvement	Expected Capacity Factor Improvement, %
Stress corrosion	Immune materials	2.3
Fuel preconditioning	Reduced rates	1.7
Thermal limits	Improved thermal limits	0.4
Nozzle cracking	Improved nozzles and spargers	1.4
MSIV leakage	Improved tooling	0.4
Refueling	Improved designs and equipment	1.5
Neutron monitor replacement	Bottom entry	0.4
Turbine island	Improved equipment	3.0
Other	Other	0.9
		<u>12.0</u>

TABLE 9
BWR/6 CAPACITY FACTOR PROJECTION*

	Capacity Factor %
Operating U.S. BWRs - 1979	68
BWR/6 Design Improvements	<u>12</u>
BWR/6 Projection	80

*Tables from Reference (5)

loss. In their recent paper, Joksimovich, et al.⁽¹⁰⁾ described an approach for evaluating economic risk by adapting the probabilistic risk assessment (PRA) methods currently used in safety evaluations. Even though the investment risk methodology described therein is still under development, it could make contributions in the financial regulatory process, and in dealing with regulatory bodies. The major contribution should be in providing background analyses that aid in clarifying options for decisions which impact upon the investment risk and thus, the cost of electricity. The methodology described should be helpful in recognizing major contributors to investment risk so that strategies for protecting the large investment can be formulated within the overall requirements for protection of the public health and safety.

Starr and Whipple⁽¹¹⁾ in a recent paper nicely sum up how the interests of the public, the NRC and the utility industry should coincide to promote increased reliability and safety of nuclear plants:

"The inherently large financial self-interest of the nuclear power industry in achieving reliable and safe operation provides the Nuclear Regulatory Commission with an opportunity to shift from an adversarial to a cooperative regulatory approach. In the TMI accident, the real loser was General Public Utilities and their customers -- not the regulatory agency or Government. The TMI accident also demonstrated that regulations and formalized processes do not assure safety. What is needed is a joint result-oriented approach that strives to improve safety, rather than merely conformance to regulation. It is in the mutual interest of the public, the NRC, and the utilities to cooperate to increase the reliability of performance of nuclear plants, using the self interest of the utilities to achieve this result."

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11. "Nuclear Plant Reliability Data System Annual Reports of Cumulative System and Component Reliability," U.S. Nuclear Regulatory Commission, (various dates).
12. "Staff Discussion of Fifteen Technical Issues Listed in Attachement to November 3, 1976 Memorandum from Director, NRR to NRR Staff, "NUREG-01381 U.S. Nuclear Regulatory Commission, November 1976.
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APPENDIX A

REVIEW OF PAST AND PRESENT HARDWARE RELIABILITY PROBLEMS

In order to identify a representative sample of major reactor hardware reliability problems, a review of published plant availability data covering the period 1970-1981 was performed. This review included the NRC's Gray Book,⁽⁴⁾ EPRI reports regarding plant availability,^(1,2,5) National Electric Reliability Council (NERC) data,⁽³⁾ and other industry work in this area. Most of the nuclear plant outages attributed to hardware reliability could be classified into a number of major categories, and these are identified in Table A-1 along with their corresponding detailed specific contributors. From this list of contributors to plant outage, some of the most significant contributors were selected for further study, based on the following criteria: the capacity factor loss due to the contributor category was at least 0.5% in any of the past four years, the replacement power cost as a result of the contributor was deemed significant, and there may have been safety implications associated with the contributor (e.g., an NRC Office of Inspection and Enforcement Bulletin may have been issued regarding a generic failure that contained a requirement for licensees to shut down their plants to check and/or replace equipment). The significant contributors chosen were: Steam Generator Tubes, Reactor and Recirculation Pump Seals, Piping Supports and Snubbers, past Fuel Problems and Fuel Failures, Preconditioning Interim Operating Management Recommendation (PCIOMR) Limits, Reactor Coolant System Pipes, Reactor Coolant System Valves and Main Steam Isolation Valves (MSIV's) (taken collectively), Turbines (taken collectively), Main Transformers, and Condensate and Feedwater Systems (taken collectively). Included for comparison purposes was a set of contributors which met the selection cri-

TABLE A-1

Major Categories of Reactor Hardware Reliability
Contributors To Plant Outages

I. Steam Generators

- (a) Early tube cracking
- (b) Tube sheet cladding separation
- (c) Feedwater sparger waterhammer
- (d) Tube thinning
- (e) Flatplate tube support corrosion (denting)
- (f) Tube lane tube failures

II. Reactor Coolant and Recirculation Pumps

- (a) Pump seals
- (b) Pump shafts
- (c) Pump motor oil degradation and leakage
- (d) BWR recirculation pump speed control motor-generator sets.

III. Reactor Coolant System Valves and Main Steam Isolation Valves (MSIV's)

- (a) RCS relief and safety valves
- (b) Pressurizer spray valves
- (c) Other RCS valves
- (d) Main steam isolation valves
- (e) RCS valve leaks

IV. Fuel Problems

- (a) Fuel densification
- (b) Fuel failures
- (c) PCIOMR limits
- (d) Fuel power restrictions
- (e) Control rod pattern adjustments
- (f) Physics testing
- (g) Other fuel problems (includes foreign objects in fuel or reactor vessel and thermal power restrictions)

V. Reactor Coolant System Pipe and Reactor Vessels

- (a) Vibration-induced failures
- (b) PWR vibration problems
 - (i) Thermal shield vibration in reactor vessel at Westinghouse plants
 - (ii) Vibration of control rods in guide tubes and vibration-related problems with vessel internals holddown spring at Combustion Engineering plants

TABLE A-1 (continued)

(iii) Vibration problems with the surveillance specimen holder tube and the burnable poison rod assembly at B&W plants

(c) BWR vibration problems

- (i) Poison curtains
- (ii) Feedwater spargers
- (iii) Jet pump supports
- (iv) Local power range monitors (LPRM's)
- (v) Thermal cycling-induced failures (feedwater nozzles and control rod drive return nozzles)
- (vi) Intergranular stress corrosion cracking

(d) Other reactor coolant system pipe and reactor vessel problems

VI. Control Rods And Control Rod Drives

(a) Electrical shorts in stator winding endturns at B&W plants

(b) Seal problems and rod stepping logic power supply failures at CE plants

(c) Plugging of filter screens within the drive assembly, failure of drives to fully insert due to leaking pistons, and cracks in control rod drive collet housings at GE plants

VII. Nuclear And Reactor Coolant System Instrumentation

- (a) Incore monitoring systems
- (b) Excore monitoring systems
- (c) Reactor coolant system pressure instrumentation
- (d) Temperature and flow instrumentation
- (e) Loose parts monitoring systems
- (f) B&W integrated control systems

VIII. Safety-Related Systems

- (a) Safety-related systems pumps, piping, valves and instrumentation and controls
- (b) Safety-related electrical systems
 - (i) Diesel generator failures
 - (ii) Gas turbine failures
- (c) Leak rate testing
- (d) BWR torus baffles and supports
- (e) Other safety-related system problems
 - (i) Vacuum pumps
 - (ii) Airlocks
 - (iii) Residual heat removal heat exchangers
 - (iv) Containment ventilation and filtration

TABLE A-1 (continued)

IX. Piping Restraints and Snubbers

X. BWR Off-Gas Systems

XI. Turbine

- (a) Blades
- (b) Controls
- (c) Valves

XII. Generator

- (a) Cooling and oil systems
- (b) Controls
- (c) Exciter
- (d) Ground faults

XIII. Condenser

- (a) Tubes
- (b) Shell

XIV. Condensate and Feedwater Systems

- (a) Feedwater pumps
- (b) Condensate pumps and condensate booster pumps
- (c) Feedwater control
- (d) Feedwater regulating valves
- (e) Other condensate and feedwater system problems

XV. Reactor Protection System

XVI. Main Transformer

terion regarding capacity factor loss prior to 1978, but having decreased losses in the last four years, below the 0.5% criterion. These include Fuel Thermal and Reactivity Limits, BWR Feedwater Sparger and Jet Pump Supports, Reactor Coolant Pumps (other than seals), Control Rod and Control Rod Drives, Generators and Condenser Tubes. These reactor hardware reliability problems are representative of those that have been solved and the effect of their resolution has been observed in the availability data.

The following sections present a brief description of each major contributor selected, the principal causes, and the corrective actions taken to solve the problem.

Steam Generators

Steam Generators have proven to be consistent problem areas for PWR plants. There has been a wide variety of problems, solutions, and operating experiences. In approximate chronological order, major problems encountered with steam generators have been:

o Early Tube Cracking

In 1970 and 1971, longitudinal cracks were found in the tubes at certain Westinghouse units. The cause was determined to be caustic intergranular stress corrosion enhanced by:

- (1) low phosphate concentrations
- (2) a reduced steam generator recirculation ratio
- (3) operation with condenser in-leakage and
- (4) operation with out-of-specification secondary water chemistry conditions.

Solutions implemented included altering the phosphate treatment.

- o Tube Sheet Cladding Separation

In 1971, at some Westinghouse units, cladding in the water box pulled away from the base metal of the tube sheet. Corrective actions were taken in all affected units. As a result, this is no longer a problem.

- o Feedwater Sparger Waterhammer

In 1973, it was discovered that at certain flow conditions the feedwater system was susceptible to waterhammer which originated within the steam generator feedwater sparger. Physical or administrative fixes were implemented, which resolved the problem.

- o Tube Thinning

Beginning in 1972, tube wall reductions were observed at certain Westinghouse and Combustion Engineering units. The tube thinning was caused by localized corrosion of Inconel-600 beneath the sludge surface near the tube sheet. A concentration of caustics in these low circulation areas was apparently present. To reduce the free-caustic condition, many units at that time switched from phosphate treatment to all volatile treatment (AVT), which reduced the sludge buildup.

o Flatplate Tube Support Corrosion

Tube denting is the only serious problem still affecting U-tube steam generators. The changeover to AVT water treatment ameliorated previous tube problems; however, AVT does not neutralize attacks of some contaminants. Corrosion of the flat carbon steel tube support plates took place, causing these plates to expand into the captured steam generator tubes. Denting and, ultimately, tube failures resulted. The corrosion of these plates is apparently aggravated by high local concentrations of chloride, ferrous, and nickel ions, as well as a locally low pH. Poor feedwater chemistry control, condenser leaks (especially at units with salt water cooling), the presence of copper in the feedwater system, and the high power densities and low recirculation ratios of modern steam generators all aggravate this tube denting problem.

The Westinghouse steam generator design, poor AVT changeover timing, and the presence of the aggravating factors have combined to make certain Westinghouse units the most affected. CE steam generators have primarily egg crate type tube supports. Therefore, they are generally less susceptible to tube denting. In addition, except at Palisades, all CE steam generators have been run exclusively, or almost exclusively, on AVT water treatment. Consequently, with the exception of Palisades, past CE steam generator performance had been very good. Unfortunately, there are some flat support plates in the operating CE steam generators. Also, a high percentage of these units are cooled by brackish or sea water. As a result, denting has been recently observed at at least one CE unit, and it is likely that there will be continued denting at CE units,

though to a lesser extent than at Westinghouse units.

o Tube Lane Tube Failures

B&W once-through steam generators (OTSG) operated until 1976 with almost no outage-causing problems. B&W had always specified high feedwater purity and AVT water treatment, and all B&W units have condensate polishing demineralizers. The failures that occurred at B&W units beginning in 1976 are the result of a new problem apparently unique to B&W steam generators. The tubes apparently failed due to high cycle fatigue at either the upper tube sheet or the first support plate. Cracks were initiated by either thermal shock from auxiliary feedwater or hydraulic shock from testing of the turbine stop valves. Vibration induced by high cross flows in the tube inspection lane promoted the crack propagation.

A second problem was discovered in 1977, which is apparently caused by tube pitting near the top tube sheet in the periphery and the open tube lane of the OTSG's. These problems have necessitated shutdowns to inspect, plug, or repair tubes, and have resulted in the imposition of more restrictive leakage limits by the NRC. However, most of the inspections and repairs have been performed during refueling outages. It is evident from the available data that steam generator problems have been, and continue to be, a major contributor to the loss of unit output. It is also apparent that the steam generator losses have been principally caused by tube-related problems (losses from non-tube problems are largely limited to the clad separation problem at Westinghouse units).

Westinghouse units have had the most steam generator losses. Some of the earlier problems (e.g., the tube sheet cladding problem, the feedwater sparger problem, and the early tube cracking problem) have been solved or else no longer cause outages, but the denting problem persists.

All PWR's are required by the NRC to do some periodic inspections of the steam generator tubes. These are usually performed during refueling operations. The five or so Westinghouse units with the most degraded steam generators have been required to perform more frequent and involved inspections. In addition, tube leakage limits have been made more restrictive. Despite that, aggregated Westinghouse losses have decreased since 1976, mainly because the routine tube inspection and plugging activities are preventing more serious unplanned outages.

Many Westinghouse units have shown very little in the way of tube thinning or denting, and most units with extensive denting have sufficient margins to permit continued operation at full power with fairly large numbers of plugged tubes. These units can continue normal or nearly normal operations while monitoring steam generator degradation and performing tube plugging as necessary.

At the Surry and Turkey Point units, however, replacing steam generators has taken up to 14 months. For most units, this would probably represent the largest single outage in their lifetime. Fortunately, this operation will not be required at most PWR's, and is not very imminent at most of the units with degraded steam generators.

Beside tube inspections and steam generator replacements, the following actions are being taken to mitigate the various steam generator tube problems:

- o Tube plugging is performed on degraded tubes as a part of a preventive plugging program.
- o Tube unplugging and sleeving in the degraded part of the tube is being done to increase the number of operable tubes.
- o "Hard spot" constraints are being installed on some tube sheets to prevent further cracking and disintegration.
- o Better feedwater chemistry control is being exercised. This involves tighter limits on condenser leak rates (and more frequent shutdowns to repair leaks), retubing some condensers with a more leak resistant material, and the installation of feedwater demineralization systems.
- o Some steam generators are being chemically cleaned. In the most degraded steam generators, however, chemical cleaning may be imprudent.
- o Other utilities with severely degraded steam generator tubes are considering replacement of the steam generators.

Obviously, not all of these corrective actions are being performed at all units. Still, some of these actions will be required at almost every operating Westinghouse and CE unit, and losses due to the steam generator tube problems will continue to remain high. Losses from non-tube related problems were largely limited to clad separation problems at Westinghouse units. No further losses are anticipated because of this problem.

Reactor Coolant and Recirculation Pumps

Of the various problems associated with recirculation and reactor coolant pumps, seal failures have been the most consequential. Seals on both of these pump types act against reactor coolant system pressure and are very sensitive to installation and maintenance practices and impurities in the seal injection water.

Generally, work on the pump seals is performed on a repair-on-demand basis. Seal lifetimes have varied greatly, and many plants make seal replacements only after a failure has occurred. These failures frequently take place during reactor startups when seal flow may be difficult to maintain and seal injection water may be dirty. A lengthy outage frequently results because the reactor must then be cooled down and depressurized.

The average equivalent full-power hours (EFPH's) lost due to pump seal problems has been higher at PWR's than at BWR's. One possible reason for this fact is that pump seals on BWR units consist of cartridges that can be replaced without removing the motor from the recirculation pump. At PWR's, the reactor coolant pump motor must sometimes be removed before the seal package can be replaced.

Both PWR and BWR units have reported outages caused by seal failures at about half of the operating units. These outages have occurred at about the same frequency per year of operating experience; however, the trends of these seal problems with commercial age have differed for the various pump manufacturers (Westinghouse, Byron-Jackson, and Bingham).

Of the three PWR vendors, B&W units have experienced the largest losses due to pump seal failures. Although such problems have occurred mostly at B&W units with Bingham pumps, pump seals at other B&W units with Byron-Jackson and Westinghouse pumps have also performed relatively poorly.

Losses from seal problems at PWR units with Westinghouse pumps have not decreased as the units have matured. Instead, the losses have tended to remain relatively constant. In contrast, losses from seal problems at units with Byron-Jackson

pumps have been high during the early life of the units, but have decreased as the units have matured. Due to a design deficiency, initial performance of Bingham reactor coolant pump seals at PWR units was poor. After a design modification, all of the problem seals were replaced during outages in 1976. This action has apparently corrected the problem.

Most of the GE BWR units have Byron-Jackson recirculation pumps. Seal failures were experienced on these pumps early in their operating history. In 1973, GE began implementing a design modification intended to improve pump seal performance. A seal purge injection system was backfitted or installed in almost all of the affected units. Because the modification was made at different times in the commercial lives of the different units, the losses as a function of age have not been consistently decreasing. Nevertheless, there has been a definite improvement in seal performance, and the losses for the first three years of commercial operation are consistently higher than that for the subsequent years. Furthermore, of the sixteen BWR units with Byron-Jackson pumps five of the six BWR-3 units have reported outages caused by seal problems, while none of the generally newer BWR-4 units that are operating with the seal purge injection system have reported such losses.

The seal purge injection system was also installed on GE BWR units with Bingham pumps. Despite that, Bingham recirculation pump seals have had poor, albeit, limited experience (only a few BWR units have Bingham pumps). Despite the decreasing loss trend for these pump seals indicated by the data, seal failures apparently continue to occur at these units.

Other problems with reactor coolant and recirculation pumps have included oil degradation and oil leakage from pump motors

at PWR units with Westinghouse reactor coolant pump motors. These problems necessitated shutdowns to inspect and replace the oil. Other units have reported problems with excessive pump vibration. At GE BWR units, motor-generator (MG) sets, which control the speed of the recirculation pumps, have been a common and persistent source of problems.

Fuel Densification

Fuel densification was first identified in 1972. Although the problems principally affected PWR's at first, the subsequent analyses of fuel densification led to operating limits on Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for BWR units. Nearly every PWR unit that was operating in 1972 and 1973 was derated, pending the completion of fuel densification studies, or until the cores could be supplied with pre-pressurized fuel assemblies.

Fuel Failures

Unlike fuel densification, fuel failures have been a larger problem at BWR's than at PWR's. Although a number of PWR units have experienced fuel leaks, the losses for this group of nuclear units are limited almost entirely to one unit. The early CE fuel design used in this unit had hydriding problems that resulted in fuel failures.

At BWR's, fuel failure problems have been much more widespread. The fuel failures at GE BWR's were mostly caused by internal hydriding of the zircaloy cladding and by pellet-clad interactions. This latter phenomenon was a consequence of rapid local power spiking following control rod withdrawals. Fission-product-enhanced intergranular stress corrosion cracking of the clad resulted. To reduce this effect, Preconditioning Interim Operating Management Recommendation (PCIOMR) limits were established. The losses that resulted are discussed below.

PCIOMR Limits

Basically, the implementation of PCIOMR limits precludes the rapid return to high power levels at BWR units following outages. Every GE BWR unit has experienced losses due to PCIOMR restrictions in virtually every commercial year since their introduction. Obviously, every major outage at a BWR unit results in losses due to PCIOMR limits. These losses, however, are decreasing as new, less conservative limits are implemented and as more of the 7 X 7 fuel is replaced by lower duty 8 X 8 fuel.

Fuel Power Restrictions

BWR fuel-related limits are difficult to quantify, partly because of the limited level of reporting, and partly because several different limits may all have been interacting simultaneously to determine the power level at which the unit was operating. Preconditioning limits, which were addressed in the preceding section, restrict the speed at which the plant can return to high power levels. Control rod adjustments (to permit even burnup of the fuel) required power reductions on a periodic basis. Finally, there are a number of safety-related limits such as MCPR (Minimum Critical Power Ratio), MAPLHGR (Maximum Average Planar Linear Heat Generation Rate), xenon restriction, peaking factors as they relate to rod block set-points, and end-of-cycle scram reactivity that restrict maximum power levels and cause reduced power operation.

Pipe Restraints and Snubbers

Losses from problems with pipe restraints and snubbers have been a continuing contributor to outages. In 1973, a design problem was discovered with degraded seals and leaking damping fluid in some hydraulic seismic restraints (i.e. snubbers). Inspections of the snubbers and replacement of the seals

accounted for large losses in 1973 through 1975. Since 1975, more of these inspections were performed as noncurtailing work and, therefore, contributed less to unit losses. However, during 1978 and 1979, the contribution due to pipe restraints and snubbers showed an increase of an order of magnitude over previous years.

Turbine

Most of the nuclear units have either Westinghouse or General Electric turbine-generators. The experience for these two manufacturers of turbines in those nuclear units is similar. Losses, however, have differed for the two types of turbines. At units with Westinghouse turbines, blade problems have had a major impact on unit performance.

Since both fossil and nuclear units have turbines and both types of plants have experienced comparable turbine problems, no extensive discussion of losses due to turbine problems will be made in this report. In general, the effect of blade failures on turbines at fossil units has been smaller than that at nuclear units, but a much higher incidence of problems with bearings and miscellaneous problems has resulted in the overall losses due to turbines at fossil units to actually be larger than at nuclear units.

Condensate and Feedwater System

Many components and problems have contributed to the condensate and feedwater system losses. The problems with feedwater regulating valves, feedwater control, feedwater pumps and condensate pumps are examined in more detail in the following sections. The remaining losses for the condensate and feed-

water system may be characterized as follows: miscellaneous pipe problems (principally leaks), feedwater heater problems (approximately two-thirds of these feedwater heater problems were tube related), problems with chemistry control and with demineralizers, valve leaks (excluding those in feedwater regulating valves), and other miscellaneous problems.

Feedwater Pumps

The data for outages caused by feedwater pump problems exhibit decreasing losses with increasing experience. This trend is evident out to the fifth commercial year. Thereafter, the results are less consistent as the limited amount of experience and atypical pump performance at single units bias the data. In the first five commercial years of data, however, the feedwater pump losses, grouped both by pump manufacturer and aggregated for the entire industry, show a definite learning curve improvement trend.

Four of the five types of feedwater pumps (Pacific Pumps, Byron Jackson, Ingersoll-Rand, and Delaval) in the data examined have performed very similarly in the first five commercial years. Losses attributed to these pumps in this time period are also distributed fairly evenly among all the units with these pumps. Worthington pumps, however, have apparently had more problems than the other pumps. Much of their associated losses, which occurred largely in the first two commercial years, resulted from pump casing cracks and diaphragm failures at two PWR units.

Basically, most PWR and BWR units experienced relatively frequent feedwater pump problems, but the average duration of the outage was usually short. In fact, about half of the incidents resulted in partial outages. In most cases, the unit was not shut down, but continued operation at a reduced power level using another feedwater pump(s) while the failed pump was repaired.

Condensate Pumps and Condensate Booster Pumps

Condensate pump and condensate booster pump problems have had little impact on unit performance. Most of the outages reported did not result in a shutdown of the unit. Losses for all pump manufacturers are consistently small, with little trend to the data.

Feedwater Control Systems

Losses from feedwater control problems differ substantially with the different reactor types. BWR losses are about half that of PWR's in data from recent years. One reason for this difference is that the feedwater control system in PWR's regulates the water levels in a number of steam generators, while at BWR's it only controls the level in the reactor vessel. Thus, the feedwater control system in BWR's inherently performs a more simple task and is likely to cause fewer outages.

There is also a disparity in the losses due to feedwater control problems between the various PWR units grouped by NSSS vendor. Westinghouse units on the average have experienced large losses, although these losses have tended to decrease substantially in the recent data. CE units exhibit very small losses in the pre-July 1976 data, but these losses have since increased. Losses at B&W units have remained uniformly small. A likely reason for the consistently small losses at B&W units follows from the fact that the level in a once-through steam generator is characteristically more difficult to control than that in a U-tube steam generator. Consequently, B&W designed a sophisticated, integrated control system that has apparently caused fewer outages.

Losses from feedwater control problems have resulted from uniformly short outages, 96% of which included a brief unit

trip. Generally no one unit has experienced disproportionate losses when compared with other units in the same class of plants. These outages, however, have been more widespread and frequent at Westinghouse units.

A large portion of the loss-causing problems with feedwater control systems at Westinghouse and CE units have occurred during startups when the system is under manual control. Startup is generally a hectic time, and manually controlling the feedwater flow can be a difficult task for the unit operators, particularly at those Westinghouse units with three or four steam generators. Losses, however, have decreased as the operators have acquired experience, and the resulting learning curve is reflected in the data for Westinghouse feedwater control systems.

Feedwater Regulating Valves

Most of these outages were full outages; but some of the problems did not require a unit shutdown. Like feedwater control problems, feedwater regulating valve problems were of less consequence at B&W and GE BWR-4 units. Also, feedwater regulating valve problems at Westinghouse units do exhibit a decreasing trend with increasing commercial age; however, there is generally no such trend when the data are aggregated for all units.

Other Condensate and Feedwater System Problems

Losses from other condensate and feed-water systems problems were headed by piping problems, feedwater heater tube leaks, and system chemistry and demineralizer problems. As a whole, losses due to this category were greater than those associated with the condensate and feedwater systems described above.

Summary

Table A-2 summarizes the selected set of significant contributors to plant capacity losses, the current status of each contributor, the cause of the contributor, the resolution proposed and/or implemented for each problem, and the effectiveness of the resolution.

References

1. "Nuclear Unit Operating Experience - 1978 and 1979 Update," EPRI NP-2092, October 1981.
2. "Major Outage Trends in Light Water Reactors," EPRI NP-755, April 1978.
3. "Ten Year Review 1971-1980 Report on Equipment Availability," National Electric Reliability Council.
4. "Licensed Operating Reactors Status Summary Report," NUREG-0020, U.S. Nuclear Regulatory Commission, March 1982.
5. "Nuclear and Large Fossil Unit Operating Experience," NP-1191, S. M. Stoller Corporation, September 1979.

TABLE A-2

Summary of Significant Contributors to
Nuclear Plant Capacity Losses (Sheet 1 of 8)

Contributor	Status	Cause	Resolution	Effectiveness
STEAM GENERATORS				
Early Tube Cracking	Solved	Caustic intergranular stress corrosion enhanced by: - o low phosphate concentrations - o reduced steam generator recirculation ratio - o operation with condenser in-leakage - o operation with out-of-spec chemistry conditions	Alteration of phosphate treatment. Limits on operation with condenser in-leakage and out-of-spec chemistry conditions.	Losses decreased.
Tube Sheet Cladding Separation	Solved	Failure of bonding between cladding and tube sheet.		Losses decreased.
Feedwater Sparger Waterhammer	Solved	Rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty valve motion.	Installation of J-tube in steam generator, other steam generator modifications. Revised operating procedures.	Losses decreased.
Tube Thinning	Solved	Localized corrosion of Inconel-600 beneath the sludge surface on the tube sheet.	Switch to all volatile treatment to reduce sludge buildup.	Losses decreased.
Flatplate Tube Support Corrosion (Denting)	Ongoing	Corrosion of flat carbon steel tube support plates causing plates to expand into the captured steam generator tubes. Aggravated by - o high local concentration of chloride, ferrous, and nickel ions. - o locally low pH - o poor feedwater chemistry control - o condenser leaks (especially at salt water sites) - not affected by AVT - o presence of copper in feedwater system	- o Tube plugging performed on degraded tubes as part of preventive plugging program. - o Tube unplugging and sleeving in degraded part of tube done to increase number of operable tubes. - o "Hard spot" constraints on tube sheets removed. - o Strengthening constraints installed on some tube sheets to prevent further cracking and disintegration. - o Exercise better control of feedwater chemistry. Involves	Losses from tube problems indicate slight decrease between 7/76 and 6/78. This trend is deceptive. Losses expected to increase again from mid-1978 on.

TABLE A-2

Summary of Significant Contributors to
Nuclear Plant Capacity Losses (Sheet 2 of 8)

Contributor	Status	Cause	Resolution	Effectiveness
Flatplate Tube Support Corrosion (Denting) (Cont'd.)		o high power densities and low recirculation ratios of modern steam generators	tighter condenser leak rate limits (and more frequent shutdowns to repair leaks), retubing condensers with a more leak resistant material, and installation of feedwater demineralization system o Chemical cleaning of steam generators o Replacement of existing steam generators with those of new design	Work in progress. No US data yet, but foreign experi- ence not optimistic.
Tube Lane Tube Failures	Ongoing	High cycle fatigue at upper tube sheet of first support plate. Cracks initiated by thermal shock from auxiliary feedwater or hy- draulic shock from testing of turbine stop valves. Crack propa- gation promoted by vibration-in- duced by high cross flows in tube inspection lane.	Same as for denting.	
REACTOR COOLANT AND RECIRCULATION				
Pump Seals	Ongoing	Installation and maintenance prac- tices. Impurities in seal injec- tion water. Inadequate seal cool- ing flow. Design deficiency (in the case of Bingham RCPs at PWR units.)	Replace seals (with new design, in case of Bingham PWR pumps). Install seal purge injection system (on GE BWR units).	In cases of PWR units with Bingham or Byron-Jackson RCPs, losses de- creased. For PWR units with W units, losses have not decreased. In case of BWR units with Byron- Jackson recirculation pumps, seal performance has improved. For BWR units with Bingham R-circulation pumps it has not.
Pump Shaft Problems	Solved	Design deficiency	Replace shafts with modified design Reduced power level until shafts could be replaced	After fix was implemented, losses ceased.

TABLE A-2

Summary of Significant Contributors to
Nuclear Plant Capacity Losses (Sheet 3 of 8)

Contributor	Status	Cause	Resolution	Effectiveness
Pump Motor Oil Degradation and Oil Leakage	Solved		Shutdowns to inspect and replace oil.	Losses have tended to decrease with increasing plant age.
Problems with W Reactor Coolant Pump (RCP) Motors at B&W Units	Solved	Design	Modify affected units.	Losses have tended to decrease since fix implemented.
Recirculation Pump Speed Control M-G Sets at B&W Units	Recurring			
PIPING SUPPORTS AND SNUBBERS	Ongoing	Design problem in hydraulic snubbers--seal material incompatible with hydraulic fluid.	Replace failed seal with seal of material compatible with fluid. Implement testing/inspection program as part of Technical Specifications.	Although replacement of seals clears up problem for failed hydraulic snubbers, still have problems with mechanical snubbers. Outage time due to snubber inspection requirements is increasing.
FUEL				
Fuel Densification	Solved		Reduced-power restriction. Replace unpressurized fuel with prepressurized fuel.	Losses decreased significantly.
Fuel Failures	Solved	Internal hydriding of Zircaloy cladding. Pellet-clad interactions caused by rapid local power spiking following control rod withdrawals.	Improved manufacturing process. Preconditioning Interim Operating Management Recommendation limits established.	Losses due to fuel failures decreased. Losses due to PCIORR limits increased.
PCIORR Limits	Ongoing	Operating restriction to reduce fuel failures.	Implementation of new, less conservative operating limits. Replace 7 x 7 fuel with 8 x 8 fuel of lower duty.	No published data show effect. Should result in reduction of losses.
REACTOR COOLANT SYSTEMS PIPES	Ongoing	Intergranular stress corrosion cracking of stainless steel RCS piping.	Immediate shutdown ordered by NRC for purpose of inspections. Periodic inspections performed with crack repair where necessary.	Losses are decreasing.

TABLE A-2

Summary of Significant Contributors to
Nuclear Plant Capacity Losses (Sheet 4 of 8)

Contributor	Status	Cause	Resolution	Effectiveness
REACTOR COOLANT SYSTEM VALVES AND MSIVs				
RCS Relief and Safety Valves	Ongoing	Actuator problems. Valve leakage.	Repair and/or replace affected components as necessary. In- creased surveillance.	Losses decrease with experience.
Pressurizer Spray Valves	Ongoing	Seat leakage. Packing leaks. Operator failures. Valve sticking due to boron precipitation on stem.	Repair and/or replace as necessary.	Losses show a decreasing trend.
Other RCS Valves		Valve leakage. Operator failure. Post-maintenance assembly. Regulatory requirements for more frequent valve cycling for testing purposes aggravate the situation.		Losses show decreasing trend with commercial age.
RCS Valve Leaks	Ongoing		Repair and/or replace as necessary.	Losses tend to decrease as unit matures.
MSIVs	Solved	o Design deficiency. o Seat leakage. o Actuator problems.	o Modified design	Losses decrease with experience.
TURBINE GENERATOR				
Turbine Blade Failures	Ongoing	o Foreign object in turbine blading. o Generic vibration problem caused by harmonic coupling of L-1 and L-2 blades with the exciter-generator system.	o Increase frequency of inspection. o Replacement of last three stages of blades on both ends of low pressure turbine near- est the generator with baffles to reduce steam pressure be- fore discharge into the conden- ser until replacement blades could be provided. Unit had to operate at reduced power for this period. o Blades repaired or replaced. o Affected blade stage was welded to shroud to give it a	Data indicate losses tend to de- crease over first six years of com- mercial operation, but then losses start to increase significantly.

TABLE A-2

Summary of Significant Contributors to
Nuclear Plant Capacity Losses (Sheet 5 of 8)

Contributor	Status	Cause	Resolution	Effectiveness
Turbine Blade Failure (Cont'd.)		vibrations and high blade root stresses.	new natural frequency. Re- placed low pressure turbine rotors with redesigned rotors that reduced steady state loading in the critical blade root and steeple locations.	
		o Stress corrosion - induced cracking (W units).	o Increased frequency and modi- fied criteria of turbine inspection.	
		o Stress corrosion cracking caused by NaOH corrodent passing through the conden- sate polishing demineralizers of B&W units.	o Replaced low pressure tur- bine rotors.	
		o Water induction (GE units).	o Replacement of affected components.	
Turbine Controls	Ongoing	o Outages due to spurious trips (variety of causes) initiated by control system.		Losses tend to show a decreasing trend over first five years of commercial operation. Between that time and the seventh year, they increase to a secondary peak, then start decreasing again.
		o Shutdowns for testing control system.		
Turbine Valves	Ongoing	Testing operator error.	None required to date.	Losses less significant than other turbine problems.
Turbine Bearings	Ongoing	o Oil leaks. o Damaged bearings. o Loss of service water to oil cooler.	o Repair leaks. o Replace affected components. o Restore service water flow.	Even though outages tend to be lengthy, total losses have been small.
		o Loss of oil cooler. o Contaminated oil.	o Repair oil cooler. o Replace oil. Repair source of in-leakage.	
		o Bearing vibration.	o Replace affected components. o Modify inspection procedures.	
Other Turbine Problems		Leaks in turbine piping or casing.	Repair leaks.	Losses have been small.

TABLE A-2

Summary of Significant Contributors to
Nuclear Plant Capacity Losses (Sheet 6 of 8)

Contributor	Status	Cause	Resolution	Effectiveness
GENERATOR PROBLEMS				
Generator Instrumentation and Control	Ongoing	Not significant		Losses show decreasing trend with increasing commercial age for all generator problems.
Exciter Failure	Ongoing	Bearing problems. Various other causes.	Replacement of affected component.	
Generator Cooling and Oil Systems	Ongoing	Leaks (either coolant or oil)	Repair and/or replacement of affected component.	
Other Generator Problems	Ongoing	Ground faults		
MAIN TRANSFORMER	Ongoing	Random incidents - o Fires - o Gas buildups - o Shorts - o Electrical disturbances - o Lightning strikes	Repair and/or replace as necessary.	Little or no consistent improvement in losses apparent from data.
CONDENSATE AND FEEDWATER SYSTEM				
Feedwater Pumps	Ongoing	- o Pump casing cracks - o Diaphragm failure		Losses tend to decrease as operating experience increases.
Condensate Pumps and Condensate Booster Pumps	Ongoing			Losses exhibit a decreasing trend with increasing unit age.
Feedwater Control	Ongoing			Losses have decreased as operators have acquired experience.
Feedwater Regulating Valves	Ongoing			No trend indicated in data from all units.
Other Condensate and Feedwater System Problems	Ongoing	o Cracks and/or leaks in extraction steam, condensate, or feedwater piping.	Repair and/or replace affected components.	Insufficient information to describe trends of losses due to problems.

TABLE A-2

Summary of Significant Contributors to
Nuclear Plant Capacity Losses (Sheet 7 of 8)

Contributor	Status	Cause	Resolution	Effectiveness
BWR VIBRATION PROBLEMS				
Jet Pump Support Failures	Solved	o Inadequate design	o Replaced with new design.	o BWR-3, most affected. Losses decreased after fix was implemented.
Feedwater Sparger Failures	Solved	o Vibration	o Series of three redesigns.	o Losses increased after discovery of problem while redesigning was in progress, until 1974, when the problem was successfully resolved. Since then, losses have dropped significantly.
Poison Curtain Vibration	Solved	o High velocity flow passing through bypass flow holes in core plate caused poison curtains to vibrate against fuel channels.	o Modified core plate design. Inspect fuel channels and replace where necessary. Reduced power level until modifications could be made.	o Losses in 1973 and 1974 were very high due to problem which only affected two units. Since 1974, when problem fix was implemented, there have been no losses.
LPRM vibration-induced wear of fuel channels	Solved	o Same as poison curtain above.	o Same as poison curtain above.	o Losses in 1975 and 1976 were very high. Since 1976, when fixes were implemented, there have been no losses.
CONTROL ROD AND CONTROL ROD DRIVE PROBLEMS				
At B&W Units	Solved	o Epoxy breakdown due to incompatibility with the wire. Moisture Side by side phasing (bifilar design). Manufacturing defects.	Repair or replace faulty Control Rod Drive Motor (CRDM). Resume operation at lower power level with the dropped rod.	Losses exhibit a decreasing trend since implementation of fix.

TABLE A-2

Summary of Significant Contributors to
Nuclear Plant Capacity Losses (Sheet 8 of 8)

Contributor	Status	Cause	Resolution	Effectiveness
At CE Units CRDM Seal problems	Recurring	Seal failure	Replace failed seals	No consistent trend obvious from data.
Rod step- control logic problems	Solved	Failure of 15-volt DC power supply.	Replaced failed power supply. Installed backup power supply.	Losses indicate decreasing trend since fixes implemented.
At CE Units Plugging of filter screens within drive assembly.	Ongoing, but not signifi- cant.	Buildup of iron oxide and other reactor crud.	Electronic lock-out of affected drive with control rod inserted.	Losses have never been significant.
Failure of drives to fully insert.	Ongoing, but not signifi- cant.	Leaking stop pistons.	Unit operation within Tech. Spec. limits until next major outage when inoperable drive can be repaired as a parallel outage activity. Changed materials.	
Cracks in control rod drive collet housings.	Ongoing, but not signifi- cant.			
Installation of control rods.	Solved	Poor quality control during control rod manufacturing.	Regulatory requirement for shutdown margin testing. Oper- ation after tests permitted only to next extended outage when affected rods could be replaced.	Problem caused major losses in 1973. After implementation of fixes in 1974, losses essen- tially ceased.
CONDENSER TUBES	Solved	Materials degradation due to corrosion.	Reduction of unit output while failed tubes are repaired or plugged. Retube condenser.	Losses show decrease compared with four years of commercial operation.

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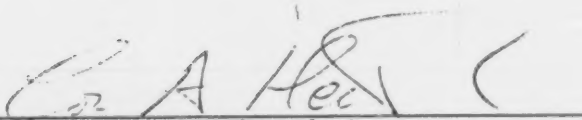
REACTOR QUALITY

Factors Affecting Managerial and Institutional
Competence of Utility Nuclear Operations

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1.0 OVERVIEW AND SUMMARY

1.1 Overview

There are a number of factors that substantially influence the quality of design, construction and operation of nuclear power plants by electric utilities. Some of these are widely understood to be different among different utilities, while many others are generic in their applicability. This paper will outline these important factors and discuss recent trends that are bringing about change in them. Factors that might be subject to influence and change if certain public policies were implemented will be emphasized, with the goal of illuminating where substantial leverage exists to modify current practice.

Any discussion of factors influencing reactor operation should begin with a brief overview of the nature of utility management today, and the reasons why the subject of management should be an important one in the overall context of discussions of nuclear power's viability in the United States.

The best way to describe present opinions on the strength of utility management today is to note that there is a strong consensus that it requires upgrading, attention, and nurturing. It is widely recognized that the nuclear reactor industry is in serious trouble today; there are no new orders for reactors, the reliability of existing reactors could be improved with substantial benefits to the utilities, the construction schedules and costs of many units now being built have shown great delays and overruns, and there are occasional serious lapses in areas such as quality control, quality assurance, and managerial accountability. Students of the problem generally agree that in many instances the underlying problem is one of managerial competence and control; specifically, all too many of the specific events that comprise the

broad mosaic of 'troubles' being experienced can be so characterized.

It would be unfair, however, to blame 'all' or even 'most' of the difficulties now being experienced in the industry to utility management. Such a one-parameter explanation is too simplistic, ignores other key contributors such as regulatory inadequacies and financial constraints, and begs more questions than it answers. Since our goal here is to identify areas where improvements can be made, it is in any event not very useful to talk about 'blame'. Rather, it is our purpose to indicate where problems lie, what might be done about them, and how widespread the difficulties are. It is important to realize that in many arenas, utility managers have been doing an excellent job. Nevertheless, there are problems today with utility management, and they are amenable to analysis and remedial action.

An overview of the utility reactor industry today reveals that there are many utilities in which competent management is carrying out its mission effectively: reactor construction schedules and costs are reasonable, management control is effective, and operational reliability of the units is quite good. However, all too many other utilities suffer from problems that are very costly in terms not only of public confidence, but in dollars-and-cents terms.

Why is there such a spectrum of performance?

Perhaps the simplest explanation is that reactor management is a highly complex, technically demanding, unforgiving discipline. That this is true is unarguable. More important, it is a relatively new area for many utilities, and one that requires the utmost attention, from the top down. Especially for those utilities which own just one reactor unit, or whose first unit(s) are newly completed or still under construction,

the complexity of the task (indeed, its sheer enormity!) has unfortunately been underestimated all too often.

This simple explanation does not capture the entire spectrum of issues in utility management today, since some difficulties seem unexplainable by it. However, it is a reasonable approximation, and as a point of departure it can be used to understand (and perhaps to remedy) a very large part of the problem.

The reasons why reactor management is so different, in a qualitative sense, from management of other electric utility enterprises are numerous. Even though other utility activities are very complex themselves (for example, a large coal-fired generating station or an intricate metropolitan substation-distribution network), there are several reasons why nuclear generating station construction and operation are very different.

An example is the atmosphere of public awareness and interference that accompanies construction and operation of a nuclear generating station. Is a very minor three-hour downtime in an oil-fired unit carried on the local radio or described on the front pages of the local newspapers? Does a minor occupational accident in which a single worker is perhaps only mildly and temporarily hurt given state-wide attention when it happens at a hydro site? Is the siting process itself dragged out in the public consciousness for years and years for any other type of facility? Do pickets and fence-climbing protesters ever oppose a natural-gas-fired electric station, either in the US or overseas? Of course not, but these routinely happen in the nuclear reactor arena, and cause a severe problem of morale, tension, we-vs.-they divisions, and ultimately a siege mentality at many nuclear generating stations.

Secondly, there is an extreme over-bureaucratization of every aspect of reactor construction and operation. More people seem to be writing more things down in more logbooks around a reactor facility than in any other endeavor known to man at least, this seems to be the case, although a few other endeavors such as the NASA manned-space-flight program were comparable in this regard. All of this bureaucratization is in the name of 'safety' or 'safeguards', often both. While it has been understood and accepted as one of the economic and organizational prices that must be paid for the benefits of nuclear-generated electricity, it has continually plagued efforts for direct, understandable management.

Thirdly, emphasis on 'meeting regulations' has been seen in some circles as somehow more important than 'safety' itself, either because it was thought that safety would inevitably flow from the adherence to regulation, or because the economic stakes were seen as so very high that avoiding 'infractions' was given top priority.

Again, as with the problem of achieving effective operations in an open public goldfish bowl, the solution to the management problems of bureaucratization and of blind regulatory adherence has not been found by at least some of the nuclear utilities.

A fourth explanation for some of the utilities' management failures has been the sheer technological complexities themselves. Many utilities that ordered their first reactor units were unprepared for the problems of staffing and managing the construction jobsite or the operating reactor once built, and turned for help to the vendors or architect-engineering (A/E)-construction firms to provide the crucial boost from inexperience to experience. For some, the reactor station would be the largest single unit in their network, and in a few cases

almost the only large unit in their network. Many utilities successfully overcame the learning gap quickly and effectively, including instituting their own internal system for finding and correcting problems as they arose, and insisting on careful analysis of experience elsewhere in operations and management. However, other utilities did not overcome the learning gap very well as time passed, and perhaps have still not done so today. This observation is best evidenced by the extent to which some utilities have continued to lean on their vendors and A/E contractors for technical support right through to the end of their operating-license interactions with NRC. At a stage when the licensee's staff should be able to answer certain questions directly and fully themselves, they still have sometimes required contractor assistance.

Yet a fifth explanation of some utilities' management ineptitude has been the peculiar semi-public, non-competitive nature of the electric industry itself. Often severely subservient to state public service commissions, often financially strapped in recent years by lags in regulatory rate relief and cost disallowances, some utilities have simply lacked the wherewithal (or thought they lacked it) to accomplish all that was managerially necessary to build and operate their reactor stations well enough. With an absolute requirement to provide all of the electricity demanded in their service areas, and insufficient resources to accomplish this task with adequate margin, and above all with a highly uncertain ability to project electricity demand adequately for planning purposes, it is no wonder that utility management has had a difficult time coping with their global needs over the past decade. This class of problems goes far beyond the area of reactor-electric generation, of course, but has exacerbated the problems of reactor owners on top of all the other problems cited above!

The preceding paragraphs might sound like the familiar litany of complaints heard within the industry, and hence dismissable

as exaggeration, self-pity, or self-service. Especially in the light of the remarkable success of a significant number of nuclear utilities, the skeptic must wonder how it is that some have done so well while others have been afflicted with (indeed, perhaps have succumbed to) the difficulties. To this there is not any single easy answer, since the issues are so intermixed. We will address the issue by exploring how recent activities have recognized and are now solving the problems. Through exploring the recent significant progress in recognition of and solution to the problems, a light can be cast on how it has come about. Furthermore, insight can be gained into whether recent trends and actions are sufficient (indeed, whether the problems cited were ever actually as serious as some perceived).

The full recognition of the problems with utility management seems to have occurred, of course, almost overnight in the aftermath of the accident at Three Mile Island (TMI). The obvious financial peril that the TMI accident imposed on TMI's operator-owner was by itself a sufficient motivation for introspection and analysis. However, in actual fact there had been a growing body of evidence about some of the problems for at least several years before the TMI accident.

Perhaps the most important evidence was the results of the Reactor Safety Study (WASH-1400) carried out by Rasmussen and colleagues in 1973-75. To the perceptive reader, WASH-1400 taught the lesson that significant accidents were not 'impossible' or 'inconceivable', especially if one considered accidents that would cause major financial loss to the utility while causing no damage to public health or property. WASH-1400 also taught the observant that human error, ineptitude in maintenance, and the like were major contributors to the accidents that dominated the risk, risk to utility property every bit as much as risk to the public. But the most

important lesson of WASH-1400 was that 'regulation' could not be equated with 'safety'. The accidents such as large LOCA's that were of highest concern to the regulators, that provided the design basis for much mitigation or prevention hardware and procedures, that had been the cause of such loud public debate and concern these accidents were found by Rasmussen to be much less important than a whole spectrum of others that the regulators had almost entirely ignored! If this were true, how could compliance with regulation be ipso facto adequate for protection of either the public or the utility's investment?

Of course, one answer given to explain the complacency of some utility managers is that, although there were some lessons from WASH-1400, the overall risk still seemed too small to worry about. Hadn't reactors been designed originally by some of the very finest scientific and engineering talent that had ever been brought to bear on any problem; and hadn't the extraordinary efforts at QC and QA assured that design and construction errors at the detailed level would be nearly negligible; and hadn't senior government officials told everybody so?

Besides, it appeared to the utility owners and their contractors that all of the issues that had been hammered on for so many years by the anti-nuclear activists were, in fact, unimportant. It seemed to some owners that these activists had generally lacked much technical know-how and their presence as gadflies or worse in the public arena served more to strengthen the resolve of utility managers that their work was proper than to cause significant introspection.

And then, with the incident at Three Mile Island, utility managers suddenly realized that weaknesses in the present system might bring down the entire endeavor; more important, that several clear paths toward remedial improvements were acces-

sible, indeed imperative. And the realization was followed by action--such as the founding of the Nuclear Safety Analysis Center (NSAC) and the Institute of Nuclear Power Operation (INPO), the strengthening of owner-vendor-AE interaction and support, and the emphasis on human factors issues.

Perhaps the archetype of the entire community's response to TMI was its sudden interest in analysis of operating data and abnormal events. Neither the industry nor the regulators had previously paid much attention to information exchange, but now they were calling for strong, effective, widely-disseminated analysis of events large and small that might carry with them any safety lessons. And this resolve has been followed by effective action. Both NRC and the industry now have institutionalized such analysis.

Another archetypical response has been the realization by all that regulation was not realistically geared to the actual problems with reactor safety. The strong emphasis since TMI on understanding the variety of human factors issues, on analyzing transients and small LOCAs, and on understanding how reactors actually behave in accidents beyond the current design basis attests to a more realistic acknowledgement of where the actual issues probably lie.

Yet another response is embodied in the last sentence of the paragraph above, in which the word 'probably' appears. Since we cannot 'know' what the future holds, it is now widely understood that it is crucial to work out as best we can what will 'probably' happen, so as to concentrate attention where it is 'probably' going to do the most good. Hence we have witnessed the advent of widespread probabilistic analysis of systems, sequences, and component failures. This in turn has led to the concentration of expertise and funds on realistic rather than conservative, design-basis analysis. This new

emphasis on realistic analysis has forced many utilities to upgrade their in-house engineering talent so that there is improved staff understanding of safety issues. This is in itself a major advance, to which the advent of probabilistic analysis has contributed in a key way.

But the most crucial change in perspective has been the comprehension among utility managers that investment protection requires a collection of activities, both within their own utility organizations and through collective action, that is highly cost-effective in an expected-value sense. This lesson of TMI, above all others, carries with it the best promise that in the future the utility management problems of the past will be gradually but firmly overcome. Investment protection now enables a utility to place specific retrofit decisions in a more realistic light. It casts the correct light on how much to spend, and how urgently, to correct a specific difficulty. It allows utility management to bring a specific monetary figure of merit (although an approximate one in actual practice) to decisionmaking. Most important is that, for most decisions, it is the correct figure of merit, since its use forces judgments into the arena of balancing costs, benefits, and risks with at least an attempt at rational comparison.

To see how much perceptions have changed, one need merely return to the days in the mid-1970s when utility resistance to retrofits was often based on claiming that reactors were 'safe enough' 'otherwise how did we get our license in the first place?' Since reactors were thought to be 'safe enough', they were obviously thought to be resilient enough that the investment was adequately protected in all areas, and with large margin. The absence of realistic rather than conservative design-based analysis wasn't troublesome; and the absence of probabilistic analysis also wasn't a problem, be-

cause the design basis itself was widely perceived as more than adequately conservative. Another manifestation of this high over-confidence was the lack of analysis of operational events. While many important individuals understood the importance of such analysis, all too many in the industry had apparently believed that 'accidents couldn't happen.' While this explanation seems quite unsatisfying, what other explanation is there for the games that were played, for example within the AEC-NRC, as to where the responsibility for such operational analysis should reside within the agency? Such turf battling stood in the way of appropriate action right up to the time of TMI!

To summarize this discussion of factors, both institutional and managerial, that influence a nuclear utility to strive for high-quality nuclear operations, it is clear that the present universal perception of the importance of investment protection is the single most influential factor. Specifically, widespread awareness that their investment is not well protected without the performance of certain activities is leading the utilities to undertake an appropriate spectrum of activities. However, not every important activity is yet in place. For example, the industry's own self-policing activity through INPO is only now just reaching what might be characterized as adolescence, with full maturity some years off. Also, some utilities still possess inadequate in-house engineering expertise, and there may be an imbalance in staffing policies between nuclear and fossil-fuels generating operations. Perhaps even more important is that the regulatory apparatus, at both federal (safety regulation) and state (financial, environmental, and siting) levels, has been only slowly recognizing that the industry's survival requires considering investment protection as a crucial input to decisions taken to achieve other goals. That regulatory change has been slow is no surprise, of course, given the inherent inertia of government. That the regulatory agencies are now beginning, on all levels,

to confront this key concern, is a promising omen for the future.

It seems unlikely that very many utility top managers will soon forget the key investment-protection lesson from the TMI accident. It also seems likely that there will be a continuation of the industry-wide activities like those embodied in INPO, NSAC, and the revitalized owners' groups, with improved industry-regulatory understanding, and resulting peer pressure on any utilities that might be perceived to be faltering. Indeed, we can expect that these institutions will improve as they mature. That they are necessary has been known since the first days of commercial nuclear power. What is not known is whether these improvements will continue or if they will be needed indefinitely.

1.2 Summary of Generic Factors Affecting Reactor Quality

The quality problems seen at nuclear power plants have many roots. The major factors seem to be:

1. For most utilities, the nuclear power plant is different from everything else in their system. Their prior experience is with small or simple installations. The upper layers of utility management may never have been involved to any extent in the details of construction or operation. Suddenly, because of the complexity of the facility and the relationships with the NRC, utility commissions and intervenors, utility management is responsible for areas in which they may have had little or no previous experience.
2. For many utilities and the contractors they choose, the nuclear power plant is a one of a kind experience.

The personnel assigned may not be familiar with the complexities involved or even the technologies involved in some instances. There is a tendency to handle the nuclear plant like any other plant, not recognizing that the consequences of error are much greater.

3. A quality assurance program is sometimes not recognized or treated as a management system which is supposed to influence work being done. It is sometimes thought to be a system of after-the-fact checks. For example, consider the concept of design control. The quality assurance requirement is to define and verify design inputs before design activity is begun. This is a restatement of the management concept that a problem should be defined before work is started on the solution. A quality assurance program is written to meet a requirement but it may not be supported by management.
4. There may be no true managerial accountability. So many people may be involved in the decision making process that sometimes no one person can control or be held accountable for the activities related to the plant. At many utilities, the management is strong enough to do the job correctly despite all the cross-currents involved. In some others, they aren't and serious problems can occur. There is a tendency in U. S. industry to overcome weaknesses in line management by adding staff functions. Program managers, quality assurance managers, and configuration managers are examples of attempts to bolster weak management. The number of "checking" functions continues to grow. The alternative of stronger management and more accountability by that management should be emphasized.

The NRC contributes to the "checker" problem. First, they had the I&E inspections, then they added the inspector in every plant. Recently, in SECY-82-352, (August 20, 1982), there is a commitment to further enhance their inspection activity. Most of the activity and emphasis of this enhancement is aimed at detail rather than major concerns.

What is needed in the industry is better "doers" or more dedicated "doers," not more "checkers/inspectors." As it has been said so often, it is not possible to inspect quality into a product, it must be built in. Another truism is that inspectors cannot find all the faults.

Skilled, trained, accountable personnel at all levels are required. Management training is a necessity.

5. Although an original design can be well thought out and adequately reviewed, the final plant configuration will not be that original design. Design changes inevitably occur for many reasons. Field changes often receive only a cursory review. In some parts of the plant, pipe runs or cable tray runs are left to discretion of the crafts. Support of the pipe or trays can sometimes become inadequate. During operation, plant personnel with experience only in small fossil plants may make what they consider to be minor modifications, often not even notifying the competent design authority.
6. The necessity for interface control may not be adequately recognized. For such an extremely complex facility, much attention should be concentrated on the interaction of the many systems, not only hardware systems but paper/management systems as well. In this area, too, the problem is mainly with changes. There may not be enough consideration of how a change in one system can affect an adjacent or complimentary system.

7. The utilities often show little or no belief or commitment to NRC requirements. The perception is sometimes that it is just more paperwork. Response is often only to satisfy specific requirements rather than to react as if there might really be a problem. NRC regulations and control continue to grow in complexity and to become more detailed, while the real problem is the lack of management awareness of the overall problem.
8. The tradeoffs between financial considerations and the quality of the plant cannot be properly evaluated if the quality considerations are not understood. Some utility management does not have the sense of the real meaning of product quality and does not understand the value of a total quality program.

1.3 Summary of Specific Management and Institutional Considerations

Nuclear Plant Construction

Observation of nuclear plant construction during the 1970's indicated a pattern of "success" or "less than successful" projects. The following characteristics appear to be generally indicative of each category:

- o Successful Construction - examples: Calvert Cliffs, Duane Arnold, Quad Cities.
 - Minimum time span of construction
 - No major licensing disputes
 - No major construction flaws
 - One contractor had complete responsibility for engineering and construction

o Less-than-Successful Construction examples: Salem, Fermi II, Shoreham, Sequoyah.

- Extensive time span of construction (i.e., 10 years)
- Challenges by intervenors or disputes with NRC
- Breakdown of project management or quality assurance
- Utility performed engineering and engaged multiple contractors
- Extensive cost over-runs

A tentative correlation could be considered as follows: "To achieve a successful construction project assign total responsibility to a major engineer/constructor." In other words, if all other factors were equal it is postulated that a dedicated engineer/constructor, whose project team had a real "sense of mission", could make a significant reduction in project time/cost and would produce a well engineered, constructed, and tested facility.

A corollary observation of construction projects in the post-TMI era is the "never-ending syndrome". As plants get more complex and major changes to systems and equipment occur during construction (including tear-outs), the engineer, constructor and owner appear to adopt a posture that "the project will never end". The construction state becomes the normal state. The operating staff begins to behave as if planning, preparing, testing, training, licensing and reporting are the purpose of its existence. In two observed situations, an individual has been plant manager for over ten (10) years, and neither plant has generated any electricity.

Management Factors/Quality of Operation

It is generally acknowledged that an operation (or organization) succeeds when the following attributes are present:

- o Clearly established mission or objective
- o Positive incentives for accomplishment
- o Direct channels of communication, authority, responsibility, and accountability

At some large, modern nuclear power plants, many of the foregoing attributes are not characteristic of the managerial arrangement. Conversely, it appears that very significant institutional forces sometimes tend to frustrate each of the desired attributes of mission, accomplishment, communication, authority, responsibility and accountability.

The conflict is illustrated by some of the basic "cross purpose" circumstances.

a. NRC Objective vs. Utility Objective

The objectives of the two major participants in the licensing arena are basically divergent. The utility wants to generate electricity. The NRC wants to achieve maximum safety, with the generation of electricity a lesser objective that is often lost sight of.

b. Operational Perspective vs. Programmatic Perspective

The operational perspective of running an electric generating plant is to marshal the necessary re-

sources of equipment, fuel and personnel and have them function to operate the plant. Other programs, such as safety, efficiency, research, etc., are conceived and applied to be supportive of the integrated operation of the facility.

The operation of a commercial nuclear power plant is more illustrative of the programmatic perspective. This perspective is most apparent from the objectives and actions of the NRC. The NRC seeks to achieve its objective of safety by requiring nuclear plants to institute and work within a series of programs. Typical of such programs are:

- Quality assurance
- In-service inspection
- Training
- Document Control
- Security
- Emergency planning
- Radiation protection

The NRC monitors and audits the performance of a nuclear plant on the basis of its ability to carry out the required programs. The establishment and performance of "programs" typically are not supportive of, and in many instances are at "cross purpose" with, the utility objective of generating electricity.

This "rip tide" of activity is observable at all levels in utility operations, from technicians who feel demeaned by inspectors and checkers, to supervisors who are frustrated by paperwork, to managers who appear harassed by regulatory program specialists.

In the current regulatory climate, the nuclear plants which operate best are those which have made the necessary mental, organizational and financial compromise between the desired operational perspective and the required programmatic perspective.

2.0 QUALITY ASSURANCE ISSUES

2.1 Identification of Important Quality Assurance Issues

Management Commitment (Management Understanding)

Many utilities in the United States that are building nuclear power plants are building one plant or one pair of plants. In most cases, these plants are different from and huge in comparison with the other generating facilities in their network. The management of a utility may not recognize the complexity of the facility it is having built nor understand the safety issues involved. This same management knows its needs a quality assurance program, primarily because the NRC says that it is a requirement. Management may not recognize a quality assurance program as an excellent management tool which adds to its strength and effectiveness but as a bureaucratic addition to the mountain of paper associated with a nuclear power plant. Because of this perception, management may not make a strong commitment to a quality program, but assign quality assurance to some minor organization. In many cases, management has not recognized the value of the quality assurance program, even after the plant is operating. Since there may be little management understanding of the value of a quality assurance program, there may be little commitment to it.

This lack of management commitment to quality and a quality assurance program is readily seen at the lower levels of the utility organization by contractors and subcontractors. The result is that sometimes the quality assurance program often gets only lip service. The required documentation may be prepared, but the actual work may be done only half-heartedly, or not at all. Obtaining a strong management commitment to quality is the biggest single factor in plant improvement.

Incentives for Quality - Economic Tradeoffs

Some utilities do not recognize quality assurance as a management tool which is of value. They may not relate a quality assurance program or a quality assurance organization to plant availability. They may not have a full realization of the effect of plant delays during construction or of the cost of down time at an operating nuclear plant.

Prior to their nuclear activity, utilities usually did not have one facility with such a major impact on their costs. Taking a plant down for even a long period of time did not have a significant system impact. There was also the mitigating influence of low fuel prices in the early days of nuclear power which made it possible to run the higher cost plants without large economic penalties.

In order to improve the quality and attendant availability of nuclear power plants, utility management must have a much greater awareness of the value of preventing failure by improving their quality assurance program and of the true cost of outage for these very expensive facilities.

Training and Qualification

In any industry, training of personnel to perform specific assignments has often been done by on-the-job training except in those instances where the job was perceived as extremely complex. On-the-job training is usually provided for personnel who are going to do work which requires some sort of manual dexterity. This training is sometimes informal with little or no previous planning. With the right person doing the training, it can be effective. Many times, it is not.

The NRC has endorsed industry standards which require training and qualification of inspection and test personnel, nuclear

plant operating personnel and quality assurance auditors. In addition, the ASME Code requires qualification and certification of welding and nondestructive examination personnel. Beyond these limited areas, training, qualification and certification are addressed only in general terms.

In a field as complicated as the design and construction of nuclear power plants, training and qualification should be given high priority.

Design Deficiencies

The design of nuclear power plants has been continually evolving. As a result each plant usually has its own design idiosyncracies--some caused by location and environmental factors, others as a result of product improvement. In addition, each plant is also in an almost continuous state of modification, as new problems occur, and as the NRC adds new requirements. Since each plant is different, the impact and form of each modification are different.

Thus the industry is faced with an extremely complex design, continually changing, with each plant usually one of a kind. In general, every design has some flaws. Even after hundreds of items of a design have been produced, design flaws are often still found. It is therefore highly unlikely that all design flaws will be found in nuclear power plants before they start operating, and even the most sophisticated design control system will not assure an adequate design. A major industry challenge is to identify those design problems which have an impact on safety or which can have the greatest potential for affecting plant availability.

Quality of Workmanship and Undetected Quality Problems

The quality of work done is affected by all the other factors

being discussed. It is affected by the management commitment to quality, as opposed to false economies in time and costs by cutting corners. Economic tradeoffs are often made which affect workmanship. Financial negotiations with contractors can cause a compromise with quality. The complexity of design and clarity of engineering instruction greatly affect workmanship. Personnel policies and training and attitude toward quality contribute to the quality of workmanship.

Undetected quality problems can have many root causes--one of the major ones being the human element. People are often unwilling to expose their own mistakes, and management attitudes or perceived attitudes toward employees frequently reinforce that unwillingness. Management pressures for production at the expense of careful work can be a contributing factor. Peer pressure and threats can also cause flaws in workmanship to go undetected. Another factor is that people will recognize a flaw but not perceive the seriousness or possible consequences of that flaw, and so they do not report or correct it. Also errors can be obscured by subsequent work on the same part or structure before an inspection or test is performed.

The undetected problem is, of course, the most serious problem.

QA as an Attitude, not a Discipline

In some utilities, quality assurance is an organization, not a system for improving the quality of work. When a nuclear program is started the commitments to quality assurance, required by the NRC, are made and a quality assurance organization established. That organization is instructed to carry out the implementation of the commitments made to the NRC. If the person selected to direct the quality assurance organization is a strong personality (usually not the case), quality

assurance is effective to some extent. In some utilities, quality assurance is something quality assurance people do, and has little impact on the rest of the utility organization.

There is a need for all utilities to realize that it is to their advantage to have a system which enhances the quality of their work, and that quality assurance is a management tool to assist management to assure that activities are performed correctly.

2.2 Summary of Information Sources Relating to Important Issues

There have been many documents written, conferences held and seminars presented on the subject of quality assurance and nuclear plant quality. In 1973, the NRC held a series of conferences and published a series of books related to quality during the design, construction and operational phases of a nuclear power plant. They were known at the time as the "Rainbow Books". Various NRC personnel followed up that activity with many instructional sessions and letters to influential members of the nuclear community.

Many ANSI Standards, IEEE Standards, and NRC Regulatory Guides relate to the subject of quality assurance. Some of them are:

ANSI N 18.1	Personnel Training and Qualifications
ANSI N 18.7	QA for Operating Plants
ANSI N 45.2	QA for Nuclear Construction
ANSI N 45.2.1 through ANSI N 45.2.23	Amplification of ANSI N 45.2 requirements

ANSI/ASME-NQA-1	Update of ANSI N 45.2
IEEE 323, 334 344,398,614	QA requirements related to electrical equipment
Regulatory Guides	1.8, 1.26, 1.29, 1.30, 1.33, 1.37, 1.38, 1.39, 1.54, 1.58, 1.64, 1.88, 1.94, 1.116, 1.123, 1.144 and 1.146

Every year, the American Society of Quality Control (ASQC) holds a meeting on nuclear power quality issues. In addition, many ANS meeting sessions address quality assurance. Subsequent to the Three Mile Island Accident, the Kemeny Commission issued a supplementary report on quality assurance.

Meeting transcripts of the Advisory Committee for Reactor Safeguards (ACRS) provide insights into safety/quality issues. The Notices of Violation, issued to the utilities by the NRC, provide information on the types of quality activities and problems which are currently being discussed.

A review of the Notices of Violations provides the following information on quality issues.

- a. Many notices address the problem of no procedures, inadequate procedures, or failure to follow procedures. This indicates a lack of management direction and a lack of interest or knowledge in correctly performing quality assurance functions.
- b. Several notices address failures to correct conditions previously identified by the NRC, again an indication of lack of management commitment.

- c. Some of the notices relate to improper maintenance or operating the reactor with some equipment inoperative.
- d. In two instances, falsifying documents because of personnel harassment is addressed.
- e. In at least two instances, lack of management controls is cited.

2.3 Discussion of Quality Assurance Issues

Management Commitment

Some utilities do not see that their role is central in the designing, manufacturing, and construction of a nuclear plant. They believe instead that they are buying a capacity to produce electricity. Their perceptions are different from those of an organization that has to build and sell a product. In some cases, utility management at the higher levels has been divorced from the building of a plant and its quality. In the era of smaller fossil plants, their major concerns were total capacity and reserves.

Another factor is¹ that, since utilities are regulated, their profits are a percentage of costs. Lack of availability of more efficient plants was in earlier times not generally¹ a large financial problem because a plant's fuel costs and operating costs were passed through to the consumer.

For many utilities, the first nuclear plant in their system represented half their capacity. This nuclear plant also required a complete reorientation of management. The first nuclear plants were, in many cases, turnkey plants. For this reason, utility management or technical staffs had little involvement with the plant's design and construction, or its quality. In some cases it was not until the mid-1970's, when

the NRC started to insist that the utility know more about the plants, that there was major utility involvement. However, utilities often complied with this direction only by minimum participation.

Since many utilities are building only one plant or pair of plants, they have only limited concepts of the complexities in which they are involved. Until the NRC has performed enough inspections to create management awareness of quality issues, the corporate quality program may lack substance.

To some extent, utilities also may have a favorite contractor they use for most of their work. If the contractor is also a stranger to nuclear work, nuclear quality assurance may appear to be an additional overlay of documentation, not a method of operating.

Often the words "quality assurance" have had a bad connotation in the nuclear industry. Management has to be made aware of the true meaning of a quality program and its value. It is evident that the value of a quality program is sometimes not recognized, since some utilities apply QA only to the safety-related portion of their nuclear plants and to none of their fossil plants.

The management of some utilities sometimes does not motivate the people under their supervision toward quality. There is a tendency in parts of the industry to take weaker engineers, with few accomplishments, and make them quality assurance engineers. This has a double effect: the QA organization is weaker, and the personnel in other organizations perceive that QA is not important.

Incentives for Quality - Economic Tradeoffs

Incentives for quality are sometimes not perceived by utility management. They may not recognize that a quality assurance

program which is properly executed will save them a great deal of money and time. However, management does recognize the cost of the quality assurance program. It is difficult to show that a quality assurance program has prevented any specific failure or solved any problem beforehand. It is difficult to relate plant downtime or plant availability to the presence or absence of a commitment to quality.

There is also, in the management of some utilities, no adequate realization of the very large costs associated with the downtime of the reactor, such as idle investment, or replacement power costs.

No one in the industry has established a satisfactory measure of the costs or benefits of quality. For industries that mass produce items, the cost of quality is readily measurable, but for the case of the one-of-a-kind nuclear plant, many arbitrary assumptions would have to be made to establish a similar cost of quality. However, establishment of such a figure of merit would trigger immediate acceptance of quality assurance throughout the industry.

One of the economic factors in the nuclear industry is the method for determining revenues and profits in the utility environment. The incentives for efficiency are not the same in a regulated industry as they are in a competitive environment.

Another cost which is not perceived and is hard to evaluate is the cost of a radiation leak or a problem which attracts great public attention. Except in an extreme situation such as Three Mile Island, there is often much more cost because of public opinion effects than there is because of the problem itself. A full awareness of these intangible costs would trigger more concern toward preventing even minor problems.

Quality itself becomes an economic tradeoff commodity. It is evident that plants cannot be constructed perfectly. Every utility and every contractor makes an economic judgment; that a certain amount of quality is sufficient. This judgment is influenced by many factors; pressures from outside organizations, such as the NRC, INPO, EPRI, and public service commissions; financial pressures, schedule pressures, and competitive pressures. The amount of quality provided becomes a tradeoff between cost and perceived risk. The perceived risk includes items like NRC acceptance.

Every organization, either consciously or otherwise, determines the level of quality which it considers adequate for the activity in which it is engaged. Two alternatives to upgrade the implementation of current quality program standards and requirements suggest themselves: the nuclear industry establish a uniform minimum quality assurance program, or a minimum level of quality be established by the U.S. Government. The feasibility and usefulness of either is questionable.

Training and Qualification

There are training, qualification, and certification requirements in the nuclear industry. Training for operating personnel is extensive and well structured. There are also training requirements for inspection and test personnel, for personnel performing "special process" activities, such as welding and nondestructive examination and for personnel who perform quality audits. There are no specific requirements for training or for determining the qualifications of technical engineering or management personnel.

There is a learning process in every new enterprise. For management personnel who have had no experience in the nuclear

industry, there is much to learn. The enterprise they are to manage is totally different from their previous experience. Since there is a wide disparity among individuals' interest in learning, there can be and is a wide disparity in knowledge among various management personnel. This can be a major factor in the success or failure of an organization responsible for a nuclear power plant activity.

The same can be said about an engineering staff. A utility organization with no nuclear experience that does not take steps to train its staff for the problems of nuclear technology can create many problems for itself.

Training of inspectors and test personnel can also be an area of weakness. The ANSI Standard and NRC Regulatory Guide related to inspection and testing personnel are very specific about training in inspection or testing but vague about training in the specific technical discipline. This can result in someone being certified as an inspector after being trained in inspection techniques, having no knowledge of potential underlying technical problems. An inspector with a background in welding and piping is not qualified to perform a technical inspection of electrical installations or concrete pours. Training and certification should be as specific as it is in the area of nondestructive examination (NDE). An NDE person certified to perform ultrasonic tests is not thereby permitted to take or interpret radiographs.

There should be more specific evaluation of personnel competence. People have a tendency to work to their level of capability and competence. A person with little skill in evaluating mechanical equipment will likely be limited to finding superficial problems like scratched paint or a loose name plate, rather than the more serious problems. Many times

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the lesser qualified person will find problems with the accompanying documentation rather than with the hardware itself.

The only specific training requirement for quality assurance personnel is that they should be trained to perform quality audits. In an environment where the quality assurance organization is responsible for most action related to quality, auditing is not one of their major concerns. Many quality assurance engineers are non-degreed personnel who have risen from the ranks of inspectors. In many instances, they do not have the broad background and experience to recognize problems and evaluate their severity.

Design Deficiencies

Increasing plant complexity, continuous design updates and changes increase the likelihood of design error. No design is error free. Even after extensive testing, there are flaws which may not be detected or which may not become evident for some time. One major design problem in an operating plant is that design modifications are often not reviewed to the same degree as the original design. They are usually not made by the original designer nor are they reviewed by him, and in many cases, the designer of the modification is not as qualified. Plant managers who come out of a fossil environment may make changes without consulting the headquarters engineering staff. They may not have an adequate perception of the difference between a complex nuclear and a relatively simple fossil plant.

Design requirements are sometimes established without adequate thought. Many times a design document will call out an entire specification, code or standard rather than those portions which are applicable to the design. With such generalizations, the whole requirement may be ignored. If requirements were established only to the level necessary, they would

be more likely to be followed. There are serious consequences of overrequirement.

Some plants are based on a previous plant design with a baseline established and with some design variations beyond that baseline. However, if the baseline plant is designed for a different utility, the second utility often makes assumptions about the first utility's actions which are not correct. As in many design activities, interfaces between designs and between design organizations can create many problems.

Some of the design problems in the industry are well known and will not be resolved easily. Steam generators and water quality are not just nuclear industry problems. Corrosion and leakage problems are common to all industries using boiler-like devices. The problem may be minimized, but it is not easily solved.

Undetected Quality Problems

Quality problems may be undetected because:

- o They may be seen but not recognized as a problem.
- o They may be buried in a structure with inadequate monitoring or they may be covered up by further processing.
- o There may be an unwillingness to expose one's own error.
- o The utility may not disclose a problem because of fear of the reaction of intervenors or other outside agencies.

- o There may be inadequate perception of the possible consequences of an action.
- o The design of the test program may have been inadequate.
- o There may be threats, peer pressure or management pressure.

There is a randomness of error and a randomness of the consequence of error. The same error in two different plants may have different consequences. Errors may be widespread, but some are detected and some are not.

An expert in quality control has said that any inspection will only discover 80% of the errors, and any subsequent inspections of the same item will only discover 80% of the remaining error. All the errors in a design or in manufacturing or construction cannot be found. Any system has to make allowances for some residual level of error and be established in such a way that those errors remaining are minor. Tradeoffs and risks have to be addressed and arrived at consciously. The present system leaves too much to chance.

Outside influences such as the public, intervenors, NRC and industry groups have to be educated so that their pressures are properly oriented to eliminating major risks. Too much attention to details in areas of little consequence may detract from finding problems of significance, or may result in excessive costs without commensurate benefits.

Quality Assurance as an Attitude

There is need for utility management to be educated so that they recognize that the quality assurance system is a system

of good management, not just a system that satisfies the NRC requirements. They then should train their personnel to recognize that everyone has a responsibility for the quality of the plant, that building quality into a design or a manufacturing operation is a quality assurance function. The basic tenets of detailed work planning, performing the work in accordance with a plan and obtaining appropriate review upon completion should apply to every activity.

3.0 REVIEW OF CONSTRUCTION PHASE QUALITY ASSURANCE PROGRAMS

3.1 Introduction

The quality assurance programs for three nuclear power plants, Three Mile Island - Unit 2, Arkansas Nuclear One - Unit 1 and Calvert Cliffs - Unit 1, have been selected for this review because they were all PWR's of the same vintage. The purpose of the review was to evaluate the individual attributes of the utility quality assurance programs and those of their suppliers in an effort to identify the quality characteristics that seem important for a nuclear plant to be well built and operated.

As the reviews progressed, a number of extenuating circumstances became apparent. For the selected plants, the regulatory guidance on the content and format of an acceptable quality program that existed when the plants' construction was started was extremely limited. At the time, the major regulatory emphasis was placed on the safety and analysis of principal design criteria and design bases and how performance objectives could be achieved, rather than on the quality assurance aspects to be addressed and implemented during the design, construction and operational phases. Accordingly, the initial preliminary safety analysis reports (PSAR's) of each of the three plants had very little to say about quality assurance. Quality Assurance, at times, existed only by reference to applicable codes and standards e.g., ASME, ACI, IEEE, APL, etc., which included in varying degrees quality aspects for those specific areas which they regulated, and by the identification of tests and inspections peculiar to specific systems and components. Quality interfaces and controls between each utility and its major contractors were very briefly discussed, if at all.

As the regulatory guidance evolved, so did the introduction of more and more quality details in the individual plant PSAR's. These additional details were identified in subsequent amendments to the original PSAR submittals.

While the review of the quality sections of the FSAR documents shows striking differences in several areas, the regulatory guidance evolutionary process is considered to be one of the major factors determining why the quality programs applicable to one plant were better or worse than the others. Relying primarily on personal experiences, it is the implementation of the quality requirements, in whatever form they took during the evolutionary process, that eventually determined the "better or worse" situations. A detailed assessment to determine how well each of the three utilities and their contractors implemented the stated requirements would have involved an extensive review of literally hundreds of internal program-implementing procedures and the supporting surveillance and inspection reports--an effort beyond the scope of this task. However, on the basis of personal experience, it has been observed that implementation of quality requirements is directly proportional to management's commitment to quality.

To promote a better understanding of how the evolution of regulatory guidance generally affected the selected plant quality assurance programs, Section 3.2 provides a synopsis of the iterations in regulatory guidance which occurred during their licensing reviews. As a specific example of program evolution, Section 3.3 describes the changes which the Arkansas Nuclear One, Unit - 1 program experienced. Section 3.4 provides a comparison of the FSAR quality assurance sections of all three plants, while Section 3.5 presents the overall conclusions.

Title 10, Code of Federal Regulations, Part 50.34 (10CFR50.34) in its earliest form, promulgated requirements for the development and submittal of Safety Analysis Reports (SAR's) and Preliminary Safety Analysis Reports (PSAR's) as part of the licensing process for nuclear power plants. Among the original guidance documents issued for the purpose of expanding on 10CFR50.34 was a publication entitled "A Guide for the Organization and Contents of Safety Analysis Reports", June 30, 1966. The major emphasis placed by these regulatory documents, at the time, was on the principal criteria for design of a facility and the design bases for those major systems and components significant to safety. These documents further provided a description of how it was intended that the plant would be built and operated to satisfy the principal design criteria and design bases and contained analyses that would show that plant performance objectives could be achieved and safety assured. Very little guidance was provided or emphasis placed on the quality assurance aspects to be addressed during the design, construction or operational phases. Accordingly, neither 10CFR50.34 nor the 1966 guidance document required the inclusion of a quality assurance program in the SAR documents. It was immediately following that period, that all three of the plants selected for this review became involved in the licensing process (Arkansas Nuclear One, Unit 1 - 11/67, Calvert Cliffs, Unit 1 - 1/68, Three Mile Island, Unit 2 - 4/68).

During the years immediately following the publication of the 6/30/66 guide and the submittal of the construction permit (CP) applications for the three selected plants, the regulatory staff responsible for reviewing permit and license applications found that most SAR's did not provide sufficient information to permit satisfactory completion of their reviews. The lack of information necessitated extensive numbers of specific requests to applicants for additional infor-

mation. As a result, the review process grew longer and more complicated as each new application was reviewed. For the plants involved in this discussion, the resultant effects of the question and answer exercises were a number of amendments to the SAR documents. The CP's for Arkansas Nuclear One, Unit 1, Calvert Cliffs, Unit 1 and Three Mile Island, Unit 2 were issued 12/68, 7/69, and 11/69 respectively, each review taking longer to be than the previous one.

It was during this period (1966 through 1970), that major changes were occurring in regulatory requirements and guidance; e.g., Appendix B was first published as a proposed amendment to 10CFR50. In addition, 10CFR50.34 was undergoing revisions to include, among other details, provisions for inclusion of quality program requirements in the SAR documents.

In 1970, the AEC began issuing a series of Safety Guides, and in 1971, a series of Information Guides, respectively designed to provide applicants with acceptable solutions to safety issues and information frequently omitted from applications. These latter guides served to reduce the extent of insufficient information, but apparently not sufficiently enough to eliminate the need for numerous additional information requests by the regulatory staff.

In November 1971, the AEC announced its intent to perform preliminary SAR reviews to determine whether sufficient information was included in an application before the actual licensing reviews were started. A further effort was made to ensure consistency in applications by the publication (in February 1972) of a proposed document entitled "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." After a review and comment period, the document was formally issued in October 1972 as Revision 1, thereby superseding both the proposed February 1972 issue and the aforementioned 1966 guide. With regard to quality assurance guidance, this docu-

ment was substantially more informative. The newer provisions of 10CFR50 were included; a quality assurance program meeting the requirements of Appendix B was to be addressed; and where safety guides had been issued, the applicant was required to indicate compliance. The Standard Format document (quality section) was subdivided into a section for quality assurance during design and construction and a section for the proposed quality assurance program for operating phase activities. A discussion of information needed for each of the eighteen Appendix B criteria was given in detail. Although at the point of issue of Revision 1 of the Standard Format document, construction permits for the three plants had already been issued, the new document had a direct effect on a later activity, namely, the operating license (OL) application and review.

Evolutionary changes occurring after October 1972 included the combining of the Safety and Information Guide Series into a new series, designated "Regulatory Guide Series" in December 1972. The standard format and content document of October 1972 was later incorporated into the Regulatory Guide Series and designated Regulatory Guide 1.70. During the succeeding years, thirty eight (38) 1.70.XX subseries guides were issued as the need arose to update portions of Revision 1 of the Standard Format.

Following this time period, OL's were issued for Arkansas Nuclear One, Unit 1 and Calvert Cliffs, Unit 1 in 5/74 and 7/74, respectively.

In September 1975, all of the changes identified in the 1.70.XX subseries guides were incorporated into Revision 2 to Regulatory Guide 1.70. Accordingly, the 1.70 subseries .1 through .38 were withdrawn. As with Revision 1, Revision 2 was expanded significantly in the area of quality assurance program guidance. Direct reference to the "Rainbow Book" docu-

ments, WASH-1283, -1309 and -1284 was added; discussion on the eighteen criteria of Appendix B was expanded; and, with the inclusion of the provisions previously contained in 1.70.6, 1.70.11 and 1.70.31, quality assurance in general became more widely applicable.

Concurrent with the evolution of Regulatory Guide 1.70, a series of standard review plans was being developed for the guidance of regulatory staff reviewers. In September 1975, the individual review plans were combined into a single document, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants...LWR Editions," (NUREG-75/087).

Approximately two and one half years after Revision 2 of Regulatory Guide 1.70 and the original issue of NUREG-75/087, an operating license was issued for the Three Mile Island Unit 2 plant.

Following this period, Reg Guide 1.70 was again revised (Revision 3, November 1978) to incorporate comments received regarding Revision 2; to improve consistency with and track revisions of NUREG-75/087 and to provide guidelines for identifying and a format for submitting nuclear steam supply system (NSSS) and balance-of-plant (BOP) interfaces for standard designs.

NUREG-75/087 was subsequently revised in February 1979 and reissued as NUREG-0800, which currently exists as Revision 2, July 1981.

3.3 Arkansas Nuclear One, Unit - 1 Quality Program Evolution

The initial submittal of the Arkansas Nuclear One, Unit 1 (ANO-1) PSAR, (11/67), was basically similar in content to

PSAR's for other plants of similar vintage. Because of the limited regulatory guidance on the subject of quality assurance, neither an overall utility quality assurance program nor individual contractor quality assurance programs were defined. General statements of compliance with various codes and standards were made, which by reference included quality requirements and criteria applicable to specific equipment and components, but the details of how those requirements would be implemented and enforced were vague at best. A brief description of the quality assurance program for the construction of structures was provided, but was limited in detail to what would or could be done. In that description, additional codes and standards specifically relating to the structural aspects were identified. Here again, each of the applicable codes and standards included some degree of quality requirements, but the details for implementing and enforcing those requirements were not provided.

The systems within the plant which were considered "important to safety" were discussed in detail, most of which included specific provisions for testing and/or inspection, however, the type of testing and inspection was primarily performance rather than quality related. (It is noteworthy to point out that the systems considered to be important to safety at that time were relatively few when compared to later regulatory criteria).

Plant technical specifications, which would have provided more details on quality assurance, were not submitted with the original construction permit (CP) application.

Following the initial submittal of the PSAR, the utility and the regulatory agency reviewers became involved in a series of question and response exercises which resulted in ten amendments to the PSAR over a thirteen month period prior to

the issuance of the construction permit in December 1968. While most of the amendments included additional quality assurance provisions, the final version of the PSAR at the time of CP issuance still did not include an overall quality assurance program definition.

Following the CP issuance, a number of additional PSAR amendments were issued as a result of further question and response exercises. It was not until Amendment #14, in November 1969 (approximately eleven months after CP issue) that an overall plant quality assurance program was issued. It is noted that the issuance of PSAR Amendment #14 was coincidental to the time period in which 10CFR50, Appendix B, was first proposed, as well as modifications to 10CFR.34 and other documents which expanded on SAR content and format.

The program described by Amendment 14 consisted of nine sections which are outlined as follows:

I. INTRODUCTION

- A. Objective & Identification of Quality Program
- B. Quality Control
- C. Quality Assurance
- D. Quality Surveillance
- E. Scope

II. ORGANIZATION

- A. Scope
- B. Arkansas Power & Light Company
- C. Bechtel Corporation
- D. Babcock & Wilcox Company
- E. Summary of Organization
- F. Records
- G. Audits

III. DESIGN

- A. Quality Control
- B. Quality Assurance
- C. Quality Surveillance
- D. Documentation

IV. PROCUREMENT

- A. Quality Control
- B. Quality Assurance
- C. Quality Surveillance
- D. Documentation

V. FABRICATION, SHIPPING & HANDLING

- A. Quality Control
- B. Quality Assurance
- C. Quality Surveillance
- D. Documentation

VI. CONSTRUCTION & ERECTION

- A. Quality Control
- B. Quality Assurance
- C. Quality Surveillance
- D. Documentation

VII. TEST PROGRAM

- A. General
- B. Quality Control
- C. Quality Assurance
- D. Documentation
- E. Equipment Testing Status

VIII. GENERAL

- A. Corrective Action
- B. Program Review

IX. BECHTEL QA PROGRAM

APPENDIX A--Organization Charts

Fig 1 Arkansas Nuclear One Quality Assurance Program
 Organization

Fig 2 Functional Organization Chart with stop work
 authority and quality control in vendors shops

Fig 3 Functional Organization Chart for Quality Control
 at the plant site (including stop work authority)

APPENDIX B

1. List of Applicable Equipment

The overall program did not specifically follow the proposed 10CFR50, Appendix B, criteria, however, it did present a far better definition of how the utility intended to control the quality of critical activities.

Conversely, the Bechtel QA Program (Outline Item IX) did discuss each of the eighteen Appendix B proposed criteria in much the same form as they exist today. A noticeable omission in the Amendment #14 submittal was the lack of identification of the B&W QA program.

SAR amendments through #19 were primarily related to the addition of technical provisions and clarification of previous issues raised by the regulatory agency; very limited additional quality provisions were presented. Amendment #20

of the PSAR (4/71) was the utility's application for the operating license (OL). Submitted as part of the application was the original FSAR document which included a specific quality assurance program section, renumbered, but basically the same presentation and format as that provided in Amendment #14, modified to include the B&W program. (Although the 10CFR50, Appendix B, criteria had been made mandatory some fourteen months earlier, the content and format of the utility program remained unchanged).

During the years following the initial FSAR submittal, numerous other amendments were issued. Of particular reference to quality assurance program requirements were Amendments #21 (10/71), #25 (3/72), #27 & 28 (8/72), #31 (11/72), #33 (12/72), #34 (2/73) and #36 (4/73). Each of these amendments served to upgrade the overall program as well as those of the major contractors. Here again, many of the amended sections were in direct response to the evolving regulatory guidance and/or requirements; plant technical specifications were first issued during this period, modifications to 10CFR50, Appendix A, were addressed, 10CFR50, Appendix B, requirements were incorporated, and a fuel and ISI QA program was added.

Approximately three years following the submittal of the OL application and 5½ years following the issuance of the CP, the OL was issued (5/74). In June 1974 the utility submitted an Operational Quality Assurance Manual to the regulatory agency for review and approval. The intent of the manual was to meet the requirements of 10CFR50, Appendix B, with respect to operation, maintenance, refueling, repair and modifications and in-service inspection. This document, upon approval, was to be referenced in succeeding SAR amendments.

The evolution of the ANO-1 quality assurance program goes on beyond this point, but it already appears that the entire program was developed in response to regulatory requirements.

It appears that the program definition, for the most part, was always in a catch-up situation, as was the case for most plants during that time period. Major modifications were made, not necessarily to reflect what was being accomplished, but to address the evolving regulatory guidance and the inquiries of the reviewing regulatory staff. The preceding is not to say that any specific plant was of poor quality, only that in numerous cases, the program description followed the work rather than the reverse. However, the performance of ANO-1 as measured by its cumulative load factor of 55.5 has been about the same as the worldwide PWR average of 56.4%. As previously stated, the bottom line is represented by how effectively the quality requirements were implemented and enforced, not by how well the program documents follow the regulatory guidance.

3.4 Quality Program Comparisons

The quality assurance program comparisons for the three plants selected for review were made using the individual program descriptions contained within the SAR documents at the point of submittal of OL applications. It should be noted that while there was less than six months difference between the submittal of the CP applications for the three plants, the differences in time between the submittal of OL applications spanned a period of greater than three years, (Calvert Cliffs, Unit 1 - 1/71, Arkansas Nuclear One, Unit 1 - 4/71, Three Mile Island, Unit 2 - 4/74). It is partially because of this difference in time that the overall program for the latter plant appears to be generally more comprehensive in form than either of the former plants. TMI-2 had the advantage of more detailed regulatory guidance promulgated during the mid-1971 through 1974 period. Notwithstanding this advantage, the results of the review disclosed a number of areas in which the QA programs of both the Calvert Cliffs and Arkansas Nuclear One plants appear to have excelled relative to that of the Three Mile Island program.

Review Methodology

A checklist comprised of 68 questions oriented around the 10CFR50, Appendix B, criteria was used for this review. The SAR QA program description for each of the utilities, as well as those of their major contractors (Architect-Engineer, Constructor, and NSSS supplier), were reviewed against the checklist. Judgement factors of (3)-comprehensive, (2)-adequate, (1)-marginal, and (0)-not specifically addressed, were applied for each of the 68 checklist questions. Based on the average of the combined total scores, overall conclusions were drawn. Specific observations were made by comparing the total scores for individual program elements. In fairness to the contractors for all three plants, it should be noted that the SAR documents were relatively limited with respect to the details of their programs. Since all contractors shared this disadvantage, it is believed that results of the review were not biased in any particular contractor's favor.

Overall Conclusions

The review concluded that in general, the quality assurance program definition for the Three Mile Island, Unit 2 plant ranked the highest of the three with a score of 62%, Calvert Cliffs, Unit 1 was next highest with an overall rating of 58% and Arkansas Nuclear One, Unit 1 placed last with 51%. However, it is difficult to identify any meaningful correlation between these rankings and the operating performance of the plants.

Specific Observations

A number of categories of specific observations became apparent as a result of the review. These categories are listed below individually and where possible, combined to show interactions.

Category 1: Program description weaknesses common to the majority of utilities and contractors -

- o The lack of detailed provisions for the indoctrination and training of personnel (QA program).
- o The lack of detailed provisions for the documentation and evaluation of test results (Test control).
- o The lack of detailed provisions for the determination of causes of adverse quality conditions (Corrective Action).
- o The lack of detailed provisions for the qualification of personnel, procedures and equipment (Special Processes).
- o The lack of detailed provisions for establishment and maintenance of item identification and control.

Category 2: Program description strengths common to the majority of utilities and contractors -

- o Provisions well defined for the independence of personnel performing verification actions (Organization).
- o Provisions for the establishment and execution of an inspection program well defined (Inspection).
- o Provisions well defined for the independence of inspectors (Inspection).
- o Provisions well defined for the performance of in-process inspections (Inspection).

Category 3: Additional specific program element strengths and weaknesses -

- o Organization: Adequate to comprehensive for all utilities and contractors, with the exception of the constructor for the TMI-2 plant; In that case, marginal to adequate.
- o Quality Assurance Program: Adequate to comprehensive for all programs with the following exceptions-

Weaknesses in the recognition of need for special skills, processes and equipment in the case of Arkansas Nuclear One, Unit 1, NSSS supplier and in all of the Calvert Cliff, Unit 1 programs.

Weaknesses in commitments for management review of quality assurance program status and adequacy in all of the Calvert Cliffs, Unit 1 programs.

- o Design Control: Adequate to comprehensive for the Arkansas Nuclear One, Unit 1 programs, with the exception of the NSSS supplier; marginal to adequate for the Calvert Cliffs Unit 1 plant programs; marginal to extremely weak for the TMI-2 plant utility and constructor.
- o Procurement Document Control: Marginal to adequate for the Arkansas Nuclear One programs with the exception of the NSSS supplier, which was extremely weak. Adequate for the Calvert Cliffs, Units 1 utility and contractors; adequate to comprehensive for the TMI-2 unit, with the exception of the utility. Change control in the case of TMI-2, however, was poorly defined by the contractor programs.

- o Test Control: Marginal to adequate for all utility and contractors; combined TMI-2 programs being the least comprehensive.
- o Control of Measuring & Test Equipment: Adequate for all utilities and contractors.
- o Handling, Storage & Shipping: Adequate for the Arkansas Nuclear One, Unit 1 utility and constructor, marginal for the NSSS supplier; adequate for all Calvert Cliffs, Unit 1 programs; adequate to comprehensive for the TMI-2 contractors, but poor in the utility's case.
- o Inspection Test and Operating Status: Adequate in the area of test status for all utilities and contractors, with the exception of the NSSS supplier for the Arkansas Nuclear One, Unit 1 plant; adequate in the area of operating status for Arkansas Nuclear One, Unit 1 utility and constructor and for all TMI-2 programs, but extremely weak in all of the Calvert Cliffs programs and the NSSS supplier for the Arkansas Nuclear One, Unit 1 plant.
- o Nonconforming Materials, Parts and Components: Poor to marginal in the case of all utilities and contractors.
- o Corrective Action: Marginal to comprehensive for all utilities and contractors, with the exception of the common weakness previously identified.
- o Instruction, Procedures & Drawings: Poor to adequate in the case of all utilities and contractors; TMI-2 programs being the least comprehensive.

- o Document Control: Marginal to adequate for the Arkansas Nuclear One, Unit 1 and Calvert Cliffs, Unit 1 programs; comprehensive for the TMI-2 programs.
- o Control of Purchased Material, Equipment and Services: Poor to marginal in the case of all utilities and the Arkansas Nuclear One, Unit 1 NSSS supplier; adequate to comprehensive for the remainder. TMI-2 contractor programs appeared to be the most comprehensive.
- o Identification and Control of Materials, Parts and Components: Common weaknesses previously identified.
- o Inspection: Common strengths previously identified.
- o Control of Special Processes: Common weaknesses previously identified.
- o Quality Assurance Records: Marginally defined by the TMI-2 utility and engineer programs; poorly defined by the Arkansas Nuclear One, Unit-1 contractor programs; adequate for the remainder.
- o Audits: Comprehensive for all Calvert Cliffs, Unit 1 programs; adequate for the Arkansas Unit 1 utility and constructor programs and the TMI-2 contractor programs; poor to marginal for the Arkansas Nuclear One, Unit 1 NSSS supplier and the TMI-2 utility programs.

One general statement that can be made with regard to a comparison between the US quality and workmanship and that of Japanese quality and workmanship is that there are major differences primarily because of the cultural differences between the two countries.

A great number of Americans tend to be nomadic in terms of careers and employment, and this is an accepted way of life. This is not the case in Japan as a whole, although in recent years there is some shifting toward the westernized ways. In most cases, a Japanese worker joins a company for life and his loyalty remains with that company. The company frequently provides him and his family with assistance for continued education, reduced prices at a company store, housing and transportation, medical care, work clothing and other fringe benefits. The worker, in short, becomes a family member of the company. Introducing another cultural difference is the concept of "saving face." While many Americans are able to "shrug-off" a reprimand for an error in judgement or a violation of an established requirement, a Japanese cannot. Subjecting himself to such a compromising and potentially embarrassing situation is the farthest thing from his mind. It is partially because of this awareness that the typical Japanese, regardless of his status, is more prone to conform to requirements than is his American counterpart. This acute awareness is carried over from his personal life to his "corporate family." In this case, not only is his selfesteem at stake, but also that of his "parents." Performing his work correctly, accurately, diligently and in strict accordance with requirements is a direct reflection on the company. To do otherwise would amount to him personally causing the company to lose face.

An associated factor that has a significant bearing on the typical Japanese worker's outlook, performance and quality

workmanship, relates to the general management system. A great number of Japanese companies encourage the principle of "bottom-up" management. This concept places a great deal of responsibility on mid-to-lower tier management and promotes a far greater involvement by the common worker in the prevention and resolution of problems. Conversely, many American companies manage strictly from the top-down; accordingly, most decisions involving new ideas on the resolution and prevention of problems are made without the benefit of literally hundreds of sources of valuable input from the people who actually do the work and experience the problems. In this regard, many American companies are converting to well proven Japanese methods; "Quality Circles" is one example. This aspect of bottom-up management promotes the involvement of all crafts, regardless of job status or position. Ideas for improvement or problem correction/prevention are screened by the self-elected "circle" steering committees or chairmen, and routinely submitted to upper management. Upper management, in turn, considers the suggested improvements seriously and very frequently adopts the ideas as company policy.

Other factors besides cultural differences and management practices appear to contribute dramatically to the differences in performance, productivity and end product quality. Such factors include but are not limited to more direct involvement by governmental regulatory agencies, more modernized facilities and methodologies in the case of the larger companies, and an equal management demand for quality as for schedule and budget. Other factors indirectly affecting individual performance and productivity which in turn affects quality include the lack of or ineffectiveness of organized labor unions, the low tolerances for intervenors and environmentalists and the high regard for national unity and well-being. Another aspect which is not commonly publicized is the "dual standard" situations employed by many Japanese companies; i.e., the requirements imposed on products sold

within the country versus those which are imposed on the same products which are for export. Many people do not realize that the "cheap", "low quality" Japanese products are, or may be, cheap or of low quality by design. A simple explanation is that those exported products are most probably in strict accordance with the procurement specifications of the importing company. Hence, an intentional downgrading or lack of emphasis on product quality, therefore expense, by the importer's specifications so that his resale of the product will show a greater return. Conversely, if the Japanese manufacturer sees a market for the product within the country, the product is frequently re-designed to introduce greater quality and reliability features. On a lesser scale, the opposite is the case. In this regard, however, downgrading for in-country use is usually because of less stringent governmental controls on a particular range of products.

Another factor to be considered is that in Japan there is more direct involvement by governmental agencies. Certain Japanese government agencies, including MITI, demand inspection hold points during inprocess, as well as final, manufacturer/constructor activities. The work cannot and does not proceed beyond these predetermined points. This is, or at least was, the case for nearly all nuclear components regardless of size or function. This is not unlike similar inspections by the US ASME code inspectorates, however, the extent of the inspections and authority of the inspectors on the Japanese side are far greater. The acceptance criteria insisted upon by the Japanese inspector are far more stringent and he does not have a waiver option, justifiable or otherwise, as does his American counterpart. By comparison, it is estimated that the combined ASME and NRC/IE inspections are 50% to 75% fewer than those performed by MITI.

The emphasis of management on product quality is yet another area of difference. Many American companies have established

a very comprehensive quality assurance and/or quality control program including detailed implementing procedures and instructions. Conversely, only a few Japanese companies were encountered that had so elaborate a program, if any at all. This striking difference would seem to have a negative effect on quality. However, in general, the opposite was the case--the bottom line appeared to be a reflection of the amount of emphasis on management's commitment to quality or quality awareness rather than on how comprehensive the program. In short, where problems are encountered by US companies, in all too many cases quality takes third priority to schedule and budget pressures; the quality program then becomes superficial.

A classic example lends insight to differing attitudes. On a Thursday afternoon, inconsistencies were observed in the final nondestructive tests for several sections of the component cooling system, large diameter piping. The Japanese contractor's attention was called to the problem which centered around a misinterpretation of the requirements for spot radiography. Instead of ordering additional radiography when unacceptable defects were found, the contractor's quality control group ordered repair and reradiography of the original welds. On the following day, additional spot radiography was performed with the result that more unacceptable welds were located. The problem was traced to two welders whose certification was marginal. The QC group elevated the problem to their site management who in turn took immediate action, irrespective of the extremely critical performance test schedule. On Friday evening, two busloads of welders and fitters from the home office (80 workers) arrived on site to begin the rework in supplement to the site workforce. By the following Monday morning, 22 large diameter pipe welds had been cut, repaired, refitted, rewelded, 100% reradiographed and the film prepared for review. All were found to be acceptable. Conversely a similar problem involving three deficient welds

at a US plant was still being arbitrated nearly three weeks after detection. Corrective action was taken reluctantly after the fourth week. The preceding experiences were just two of many indications that the typical Japanese commitment to quality is more than superficial. There are obviously cases where this is not true, but normally when it is recognized that errors have been made, prompt corrective action is taken and firm measures are taken to prevent recurrence. In the cases in point, the Japanese welders were taken off the job, (completely retrained and recertified; their welding record during the remainder of the construction was superior). In the other case, the time and effort spent arguing whether the welds were unacceptable or marginally acceptable led the welders to believe that their management was willing to accept inferior work to meet schedule. Although the argument was ultimately lost and repairs were made, the welding records of the individuals involved probably remained marginal for the remainder of the project; a direct reflection of the attitude of their superiors.

All of the preceding is not to be construed to be a general condemnation of the quality of American products. The points made are that on the average, Japanese management's attitude towards and commitment to quality, perhaps at some sacrifice in profit, usually exceed our own; conformance to requirements, however simple the program, is generally better at all levels.

4.0 REVIEW AND EVALUATION OF UTILITY PERFORMANCE

4.1 The NRC Inspection and Enforcement Process

The NRC Office of Inspection and Enforcement is responsible for: inspecting NRC licensees to ascertain whether they are complying with NRC requirements (regulations, rules, orders and license provisions) and to determine whether these licensees are taking appropriate actions to protect nuclear materials and facilities, the environment, and the health and safety of the public; inspecting applicants for licenses as a basis for recommending issuance or denial of a limited work authorization, construction permit, or an operating license; inspecting suppliers of safety-related services, components and equipment to determine whether they have established QA programs that meet NRC criteria; and for enforcing Commission orders, regulations, rules and license provisions.

The inspection methodology varies for different types of licensees. For certain licensees, such as utilities who will operate power reactors, the inspection process starts when the utility announces its intent to construct and operate the reactor. Other licensees, such as fuel facilities, are inspected initially just prior to facility operation. Materials licensees are inspected after operations are initiated.

The Office of Inspection and Enforcement's present authority is defined in a Proposed General Statement on Policy and Procedure for Enforcement, which was published in the Federal Register on October 7, 1980. This is intended to become Appendix C to 10CFR2 and the Proposed Statement is being used in the interim.

The policy states the goals of the NRC enforcement program as follows:

- o to ensure "compliance with NRC regulations and license conditions";
- o to obtain "prompt correction of licensee weaknesses";
- o to deter "future noncompliance through strong enforcement measures"; and
- o to encourage improvement of licensee performance, thus enhancing the degree of protection of public health and safety, common defense and security, and the environment.

The Enforcement Policy establishes penalties (i.e., monetary, license, criminal) to assure that a licensee does not benefit from being in violation of regulations.

The Policy Statement divides the severity of violations into six categories and assigns them to seven areas of concern. Severity Categories I, II and III have major regulatory concern, and are the most significant. They involve violations which have potential or actual impact upon the public. Severity Category IV violations include degradation of essential safety function systems or management control systems. Category V includes less significant violations of other than minor concern and Category VI covers violations of minor concern.

The seven areas in which I&E conducts its inspection activities are:

- 1) Reactor Operations
- 2) Facility Construction
- 3) Safeguards
- 4) Health Physics

- 5) Transportation
- 6) Fuel Cycle Operations
- 7) Materials Operations

The functions of the Office of I&E are divided into three main tasks, which are: Inspect & Investigate, Evaluate and Inform, and Enforce. A breakdown of each of these tasks follow.

A. Inspect & Investigate

- 1) Inspect licensees and their contractors and suppliers, applicants, persons, individuals and any organizations subject to NRC jurisdiction to:

B. Evaluate and Inform

- 1) Evaluate notifications of and information concerning incidents and accidents that present a potential or actual threat to public health and safety, the common defense and security, the environment, or the safeguarding of nuclear materials and facilities.
- 2) Evaluate the results of inspections, investigations, inquiries, enforcement actions, and reports by licensees and other organizations and individuals.
- 3) Inform, as appropriate, the Commission, other NRC Offices, other government agencies, licensees, and the public of the results of reviews.

C. Enforce

Assure compliance with NRC requirements including rules, regulations, orders and license provisions.

The Enforcement Actions that the commission may take following discovery of violations are:

1. Notice of Violation - This is the formal method of notifying the licensee of a non-compliance, and may be the only action taken.
2. Civil Penalty - Fines are usually imposed if a licensee commits a major violation, willfully commits a violation, or knowingly fails to report a violation. Civil penalties may be as high as \$100,000 per violation per day, with no ceiling.
3. Orders - Orders may be issued to modify, suspend, or revoke a license to protect the public health and safety.
4. Other Enforcement Actions
 - a. Enforcement Conferences - discussion between NRC and licensee management regarding regulatory problems, licensee actions, and NRC enforcement options.
 - b. Bulletins, Circulars Information Notices - notification to licensees of generic problems; calling for required actions or making recommendations.
 - c. Immediate Action Letters - confirm licensees' agreement to take corrective actions.
 - d. Referral to Department of Justice - for alleged or suspected criminal violations.

- e. Escalation of Enforcement Action - combinations of the above actions may be used in cases of repeated severe violations.

The NRC's Systematic Appraisal of Licensee's Performance, (SALP), review is conducted and headed by office-level managers from the NRC. A year's worth of data on the utility's activities is collected and reviewed focusing on items of non-compliance, and presented to the utility in a one-or two-day meeting. This type of review does not detail specific events but looks at trends.

The objectives of the review are to:

- o improve the licensee's performance,
- o provide a basis for allocation of NRC resources, and
- o improve the NRC safety program.

The following listings denote the areas that are reviewed for both operating reactors and reactors under construction.

PERFORMANCE ANALYSIS AREAS FOR OPERATING REACTORS

This would include, for instance:

- o plant operations,
- o personnel, training, and plant procedures,
- o design changes and modifications,
- o emergency preparedness,
- o security and safeguards,
- o audits, review and committee activities,
- o corrective actions and reporting.

PERFORMANCE ANALYSIS AREAS FOR REACTORS UNDER CONSTRUCTION

The areas generally reviewed would be:

- o quality assurance,
- o safety-related components
- o fire protection,
- o corrective actions and reporting,
- o procurement (including receipt and storage),
- o training.

PERFORMANCE ANALYSIS AREAS FOR REACTORS UNDER CONSTRUCTION
WHICH ARE ALSO IN THE PREOPERATION TEST PHASE

The areas generally reviewed would be:

- o plant operations preparation,
- o fuel loading preparation,
- o maintenance,
- o security and safeguards,
- o surveillance and preoperational testing,
- o emergency planning,
- o audits, reviews, and committee activities.

AREA PERFORMANCE CATEGORIZATION

The Category I areas are defined as those areas in which the appropriate attention by the licensee is apparent. The Category I area is identified by the lack of significant administrative or management control problems, or regulatory issues.

The Category II areas are defined as those areas in which an adequate amount of licensee attention is being achieved. The Category II area is identified by the presence of only minor administrative or management control problems, or regulatory issues.

The Category III areas are defined as those areas in which more licensee attention is needed in order to improve perfor-

mance. The Category III area is identified by the presence of significant administrative or management control problems, or regulatory issues.

FACILITY CATEGORIZATION

The Category I facility has little evidence of significant administrative or managerial problems and routinely exceeds nuclear safety requirements.

The Category II facility's administrative or managerial problems are less significant than those identified above, and, where they exist, detract little from the licensee's ability to meet nuclear safety requirements.

The Category III facility is evidenced by the presence of significant administrative, or managerial problems in several activity areas as indicated by:

- o substantive construction or operational events when compared to others,
- o significant violations (when compared to others),
- o evidence of repeat violations or escalated enforcement actions, or
- o several regulatory issues involving the facility's performance.

The Office of Inspection and Enforcement has established Performance Appraisal Teams, (PAT) that conduct reviews only of operating plants. The review is conducted by members of the Performance Appraisal Section, NRC Office of Inspection and Enforcement. The team reviews management controls in approximately eight areas, and then uses the findings to rank

the utility with others in the industry. These reviews are normally conducted by a team of at least five inspection specialists whose review may take two to three weeks at a licensee's facility. The objective of the inspection is to evaluate the management control that had been established in support of licensed activities. The results provide input to the NRC evaluation of licensees from a national perspective.

The inspection effort covers licensed activities in selected functional areas. In each of the functional areas the inspectors review written policies, procedures, and instructions; interview selected persons; and review selected records and documents to determine whether:

- a. The licensee had written policies, procedures, or instructions to provide management controls in the subject area;
- b. The policies, procedures, and instructions were adequate to ensure compliance with the regulatory requirements;
- c. The licensee personnel who had responsibilities in the subject areas were adequately qualified, trained, and retrained to perform their responsibilities;
- d. The individuals assigned responsibilities in the subject area understood their responsibilities; and
- e. The requirements of the subject area had been implemented and appropriately documented in accordance with management policy.

The specific findings in each area are presented as observations, which are inspection findings that the inspectors

believe to be of sufficient significance to be considered in the subsequent evaluation of the licensee's management performance. The observations include perceived strengths and weaknesses in the licensee's management controls that may not have specific regulatory requirements or guidance.

The observations provide the basis for drawing conclusions in each inspected functional area. The conclusions are presented as Above Average, Average, or Below Average, and represent the team's evaluation of the licensee's management controls in each area. Some observations identified as weaknesses may be potential enforcement findings. These observations are discussed with the licensee and are presented to the NRC Resident Inspector. The followup of these items is by the NRC Regional Office governing the licensee's activities.

To date the PAT appraisal has only resulted in one significant enforcement action where a utility was fined as a result of the PAT visit. Although these reviews are conducted in an extensive manner and with thoroughness, the value of their application has yet to be determined.

4.2 The Institute For Nuclear Power Operation (INPO) Evaluation Process

INPO has been conducting evaluations of nuclear plant site activities to make an overall determination of plant operating safety, to evaluate management systems and controls and to identify areas which may need improvement. Its evaluations are based on performance objectives and criteria and information assembled from discussions, interviews, observations and reviews of plant documents.

The performance objectives are broad in scope. Each generally covers a single, well-defined management area. The supporting criteria are more narrow in scope and typically describe a

specific activity that contributes to the achievement of a performance objective.

The criteria do not necessarily address every activity associated with the performance objective. Therefore, meeting all the criteria does not ensure that the performance objective is fully met. Conversely, it is recognized that a nuclear station may effectively achieve the performance objective without meeting each specific criterion. For these reasons, INPO's plant evaluations emphasize the performance objectives rather than focusing solely on the supporting criteria.

The criteria are results-oriented. The methods for achieving the desired results are generally not stated. Thus, considerable judgment is required in applying the criteria.

An evaluation team examines station organization, training and qualifications, operations, maintenance, radiological and chemical activities, and site technical support. Performance evaluations are based on the best practices, and not on minimum acceptable standards. Report recommendations are not limited to minimum safety requirements, but address high standards of safety and reliability in nuclear plant management and operation.

Application of recommendations in the INPO report should consider the possibility of weakness in other similar areas, and any generic weaknesses which might be identified by the utility management.

To follow the timely completion of the utility responses, INPO requests written notification upon achievement of improvements at key target dates that the utility has established.

Although INPO has conducted a number of these reviews, there is little public documentation on the reports or corrective

actions. Therefore, no comparison or correlation can be made between INPO's reviews and those made by NRC. Additionally, there does not seem to be any exchange of information between NRC and INPO to benefit the utility and, as such, conflicting or missing reviews may be inherent in the total review process, thereby detracting from an overall benefit to the industry.

4.3 Effectiveness Of Formal Review Activities

Prior to TMI and the findings of major review groups of utility management capabilities the AEC/NRC performed only a pro forma review of the utility organization. The licensing review was centered around the operating organization (qualified operators). The AEC/NRC was satisfied if the proposed utility organization had a qualified architect engineer and QA program. Guidance for the pre-TMI organizational reviews was slight. ANS 18.1 of 1971 and the standard format and content for safety evaluation reports were the two main review guidelines in use.

As examples of the results of the licensing review of the organization of nuclear plant management prior to TMI the safety evaluation reports (SER) of three proposed plants were examined:

- o A utility proposing to build a first time unit (Grand Gulf)
- o A utility with operating units proposing to build an additional unit, (Millstone Unit 3).
- o A utility proposing to build an additional unit, while its first is still under construction (Jamesport 1&2).

As was readily seen from the NRC's safety evaluation report (SER) findings, specifically their 13.1 Sections, of approximately two pages each, there was no indepth review of the organization or the qualification of personnel.

These AEC/NRC reviews are in sharp contrast to the comprehensive organizational reviews conducted after TMI-2. For example, the Commanche Peak SER, Section 13.1.1, contains nine written pages on the Corporate Organization alone, including details of the staff and their individual qualifications, and a similar section is devoted to the operating plant organization.

Both the Rogovin and Kemeny reviews of the TMI-2 accident stressed the need for a more organized method of qualified utility and regulatory involvement in the design, construction, and operation of nuclear power plants.

The NRC reacted by producing draft documents such as Task Action Plan Item I.B.1, "Management for Operations" and, NUREG/0731, "Guidelines for Utility Management Structure and Technical Resources." The Office of Inspection and Enforcement instituted the "Systematic Appraisal of Licensee Performance" (SALP), and began a program of operating plant reviews by teams of specialists, "Performance Appraisal Teams" (PAT). The industry for its part responded by instituting utility reviews of utility operations by members of INPO. As can be seen from the typical post-TMI SER, there has been a marked increase and emphasis on review of utility personnel, organization and qualifications. However, the NRC is only concentrating these post-TMI requirements on current operating license actions. There has been no effort by the NRC to apply the major findings of TMI-related studies in utility management to operating organizations, although it is clear from the Rogovin and Kemeny reports that their findings were directly applicable to all utility organizations.

Kemeny "Although the Commission considers the responsibility for safety to be with the total organization of the plant, we recommend that each nuclear power plant company have a separate safety group that reports to high-level management. Its assignment would be to evaluate regulatory procedures and general plant operations from a safety perspective; to assess quality assurance programs; and to develop continuing safety programs."

Rogovin "Strong measures to strengthen the onsite technical capability and management of utilities at reactor sites, including a new philosophy and new program for improved operator training; and new NRC requirements to ensure that qualified engineer supervisors with intimate knowledge of the plant will be part of the onsite supervisory management chain on every reactor operating shift."

Kemeny "Integration of management responsibility at all levels must be achieved consistently throughout this industry. Although there may not be a single optimal management structure for nuclear power plant operation, there must be a single accountable organization with the requisite expertise to take responsibility for the integrated management of the design, construction, operation, and emergency response function, and the organizational entities that carry them out. Without such demonstrated competence, a power plant operating company should not qualify to receive an operating license."

Rogovin "We found no evidence that the safety committees recognized the weaknesses that were developing in the Unit 2 management structure. Rather, their

attention was primarily focused on licensing and operating experience matters."

Ironically there was in existence a document, WASH-1130, Revised, "Utility Staffing and Training for Nuclear Power," June 1973, which could have provided, to both the industry and the regulatory authority, insight as to the complex needs for interactive operating organizations with many qualified individuals.

"The technical complexities, safety requirements and environmental considerations of nuclear power make it essential that top management in utilities develop general understanding of these requirements and considerations as an early stage. These include reactor technology, safety, the AEC's licensing program, the development of public understanding, environmental protection, an awareness of engineering standards and the value of quality assurance. Utility top and middle management and technical support personnel properly trained in these areas constitute the best assurance that the utility is maximizing its chances of bringing a nuclear plant on the line with a future of safe, reliable, and economic operation. "A high level of utility effort and closer attention in a management sense is necessary for nuclear power in view of its technical sophistication and regulatory requirements. The demands and complexities of nuclear power emphasize the importance of utility management's developing its own nuclear knowledge."

"Five areas of needed specialized knowledge are:

1. An understanding of reactor technology and safety considerations involved in reactor design, construction, testing, operation and maintenance in service.

2. An understanding of AEC's licensing program.
3. The importance of developing public understanding of nuclear power.
4. An understanding of how to apply engineering standards and quality assurance so as to achieve safe, reliable, and economic operation.
5. The impact of nuclear power on the environment."

One can easily see that the actions contemplated as necessary by utilities were perceived almost 10 years before the TMI accident but the exact implementation of the necessary management developments was not generally understood internally among utilities.

The only organizations which are reviewing in-depth utility operating structures are INPO and the NRC PAT teams and there has not been sufficient experience to make an assessment of their overall effectiveness.

The NRC re-organized itself to deal better with reviews of utility management and has been overseeing utility organizations for new operating licenses to a greater depth than before. However, it has been almost four years since TMI and its lessons learned, and the NRC still does not have any final guidance for utility management organizational control for design, construction and operation of nuclear plants. As a result many utilities have reverted to their passive pre-TMI roles, waiting and responding only to NRC direction.

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SURVEY AND ASSESSMENT OF THE TECHNOLOGICAL OPTIONS AVAILABLE
TO THE NUCLEAR INDUSTRY IN THE 1980 TO 2000 PERIOD

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INTRODUCTION

So many different and often conflicting ideas, positions, and claims have been advanced with respect to the safety and economics of nuclear power plants since the Three Mile Island accident that the situation has often seemed hopelessly confusing. Now, after over three years of intensive review, the situation is beginning to clear, and a number of key points have emerged. This report summarizes the results of a representative set of the more important recent studies. The first section considers criteria and figures of merit for use in this review. The next section treats the conventional light water reactors and improvements that can be made in them within the framework of the basic design concepts. The third section is concerned with the heavy water CANDU type of reactor. The fourth section is concerned with a completely new approach to the design of the light water reactor to make it as invulnerable as possible to any type of accident, sabotage, or act of terrorism. The fifth section considers the advantages from the safety standpoint of the high temperature gas-cooled reactor. The sixth and last section compares these various types of reactor and the possibilities for evolving exceptionally safe versions that also show promise of being economically attractive.

TABLE 1. SUMMARY OF DATA FOR TYPICAL POST-TMI NUCLEAR PLANT DESIGNS

	PWR (Sequoyah)	PWR- SNUPPS	PWR (Size- well B)	BWR/6 (Grand Gulf 2)	PWR KWU	BWR KWU	CANDU (Pt. Lepreau)	ASEA-ATOM PLUS	HTGR SL/C	HTGR- Modular Plant
Design power output MWt	3411	3411	3411	3833	3780	250	2181	1600	2240	250
MWe	1125	1150	1100	1302		100	633	520	858	100
Thermal efficiency, %	33.0	33.7	32.1	34	34.4		29.0	31.6	38.5	40
Reactor inlet temp. °C	285	292	309	282	291	125	266	263	318	250
Reactor outlet temp. °C	321	328		286	326	165	310	293	688	700
Reactor core dia., m	3.4	3.37	3.37	4.8	3.6	2.9	6.28	3.85	12.92	3.45
Reactor core height, m	3.66	3.65	3.65	3.81	3.9	2.3	5.94	1.97	9.45	8.0
Reactor core power density, kW/L	102		102	56	93	25	11.85	69.4	7.1	4.2
Reactor pressure, bar	155	155	155	71.7	153	7	100	90	72.4	70
No. of containment shells	1	1	2	2	2	2	1	1	1	1
Natural thermal con- vection cooling of core	No	No	No	Yes	No	Yes	Yes	Yes	No	Yes
Severe core damage factors:										
Average heat flux, W/cm ²	68.5	68.5	68.5	48.7	48.5		64	47.1	16.3	
Heat capacity closely coupled to fuel KJt/KWt-°C	0.012	0.012	0.012	.0263	0.0131		0.017 (+0.5 ln D ₂ O)	0.018 (+7.3 ln pool)		
Time to major fission product release from fuel, s	45	45	45	110	49	10 ⁴		6 × 10 ⁵	36,000	
PRA for severe core damage (per year)	110 × 10 ⁻⁶ (Ref. 2)		1.2 × 10 ⁻⁶ (Ref. 5)	10 ⁻⁶ # (Ref. 1)	10 ⁻⁵ (Ref. 11)	5 × 10 ⁻⁶ (Ref. 12)		<10 ⁻⁶ (Ref. 17)	40 × 10 ⁻⁶ (Ref. 18)	

*The PRAs listed were estimated by different groups using somewhat different methods, hence are not exactly comparable. Where a range of values was given in the estimate, the median value was listed here.

**For latest version of this type of plant.

CRITERIA

It is difficult to establish criteria for assessing the relative merits of different nuclear power plant designs. One way to attempt to quantify the nuclear safety characteristics of a plant is through probabilistic risk assessment (PRA); whatever its shortcomings (e.g., large error bands and neglect of certain external events), this approach ought at least to indicate the relative probability of a serious accident when comparing two different nuclear plant designs. For the purposes of this paper it has also seemed worthwhile to summarize some parameters affecting the probabilistic risk assessment, such as the heat capacity of the reactor core, whether or not the core afterheat can be removed by natural thermal convection, the number of containment envelopes employed, and the time that would probably elapse before severe core damage occurred in the event of a complete loss of power to both the main coolant circulating equipment and the emergency core cooling system.

A parameter of interest affecting both the capital and fuel costs is the thermal efficiency, and, closely associated with this, the peak temperature of the reactor coolant leaving the core. Some quantities affecting the consequences of an accident are the reactor design power output, the power density, the surface heat flux in the core, and the heat capacity closely coupled to the fuel. These and some other pertinent factors are summarized in Table 1 for some representative designs.

The economic viability of a nuclear plant concept is a vital consideration that depends on the safety characteristics as well as the capital cost,

availability, reliability, and difficulties of maintenance. It is exceedingly difficult to estimate these characteristics from a paper design.

There are so many site and organization dependent factors that, even for the same basic design of plant, a unit may cost three times as much when built by one contractor on one site as when built by another contractor on a different site. Similarly, marked differences in plant availability have been experienced with the same basic power plant design. Use of standardized plant designs should help reduce not only the initial capital cost, construction time, and licensing effort, but it should also help reduce maintenance costs and the incidence of forced outages, thus increasing the plant availability.

Major factors affecting cost and construction time requirements are licensing and quality control. If a new improved design is one or two orders of magnitude better from the standpoint of safety than a conventional system, and is so perceived by both the regulatory agency and the public, it would seem that major reductions in construction time, hence in capital costs could be effected, but it is difficult to assess the magnitude of such resulting reductions in costs. While there are many parameters that influence capital costs, for purposes of this report it seems prudent to limit the choice of cost parameters to the number of cubic yards of concrete and the weight of steel estimated for the nuclear portion of the plant. These parameters are not generally available, but data for some reactors are included in a table later in this report.

The difficulties to be expected in operating a nuclear power plant are extremely difficult to assess. A recent EPRI survey of eleven utilities to

get the benefit of their operating experience¹ disclosed that the 57 utility management, operating, and maintenance personnel interviewed generally felt very strongly about a number of vital points closely related to safety problems. These points are as follows:

1. NRC mandated additions of progressively more plant equipment in an effort to improve plant safety leads to such severe complications that the operability, maintainability, and availability are adversely affected, while the additional complications associated with the many superimposed systems are so great that many believe them to be counter-productive from the safety standpoint.
2. 1200-1300 MWe plants are too large.
3. Containment buildings should be larger to provide adequate space for maintenance.
4. Present plants respond too rapidly to transients.
5. Nuclear plants should be less sensitive to events in secondary systems.
6. The auxiliary feed water systems need simplification and improvement.

As will be more apparent later, all of these points have important implications with respect to the attractiveness of advanced design concepts.

LIGHT WATER REACTORS

Six light water reactor plants were selected as representative examples for discussion of the principal safety and cost problems characteristic of LWRs. The first column in Table 1 gives data for the TVA Sequoyah plant, one of the more recent PWR designs built in the 1970s.² The data for this plant have been included to provide a point of departure and basis for comparison with post-TMI designs.

Post TMI Improvements in Conventional LWRs

Perhaps the most significant development to emerge from the TMI accident is the realization that the probability of a major financial loss by the utility as a consequence of a core melt accident is much greater than the probability of any significant hazard to the general public. Thus a probabilistic risk assessment (PRA) of a partial core melt accident becomes a particularly important consideration. The prime consideration from the safety standpoint is the release of fission product activity from the fuel elements to the primary coolant system, hence overheating of the coated particle uranium carbide fuel in an HTGR would give an accident comparable in severity to melting of the fuel in an LWR. In view of this, the term PRA for severe core damage has been used in Table 1 so that PRAs for both LWRs and HTGRs could be included under the same heading.

A survey of PRA estimates for severe core damage as found in one of the IEA studies under the Second Nuclear Era Program³ have been summarized in Table 2. Inspection of this table shows that the probability of such an accident appears to differ by a factor of as much as 100 from one LWR reactor design to another. In general, in those cases where the probability of a core melt is relatively high, only one or few items were responsible. Thus a much lower probability of severe core damage could be achieved by introducing redundancy or additional capacity into particular system elements, notably the emergency core cooling system, the feed water system for the steam generators, and the diesel generators kept in standby to supply electric power to the pumps and other equipment in these systems. However, once the major weak points in a system have been corrected, the potential

TABLE 2. PRAs FOR CORE MELT ACCIDENTS IN LWRs²
 (PRAs for severe core damage; PRAs for activity release to
 atmosphere are lower by a factor of about 100)

Reactor	Description	Median (10 ⁻⁵ /RY)
Wash-1400 (72-75)		
Surry	W, 3-Loop, Dry	6
Peach Bottom	BWR-4/MK-1	2.9
RSSMAP (77-81)		
Sequoyah	W, Ice	11
Oconee-3	B&W, Dry C	7.9
Calvert Cliffs 1	CE, Dry C	200
Grand Gulf	BWR-6/MK-3	5
IREP (79-82)		
Arkansas 1	B&W, Dry	5
Crystal River 3	B&W, Dry	13
Calvert Cliffs 1	CE, Dry	200
Millstone 1	BWR/MK-1	30
Browns Ferry 1	BWR/MK-1	25
PRA (80-82)		
Yankee Rowe	W, 175 MWe, SPH	--
Big Rock PT	BWR, 71 MWe/SPH	45
IP 2	W, Dry	40*
IP 3	W, Dry	9*
Limerick	BWR-6/MK-2	2.5*
Zion 1 & 2	BWR	4.7*
Others		460
Biblis		4

*Preliminary, under critical review.

TABLE 3. VALUE OF TECHNICAL FIXES IN
 REDUCING THE PRAs OF TYPICAL LWRs

	Oconee-3 (4-Train)	CR-3 (W F&B)	Davis-Besse (W/O F&B)
Core Melt Probability	7.9×10^{-5}	13×10^{-5}	260×10^{-5}
Factor Lower if:			
Perfect AFWS	1.4	2.4	21.7
Perfect HPSI	1.1	1.1	1.0
Perfect AFWS + HPSI	1.5	2.5	21.7
Perfect Emerg. Power Supply	1.0	1.2	1.0
Perfect Feed/Bleed	1.6	2.6	52.0

for further improvements is generally small and the further individual improvements are unlikely to be cost effective.

Table 3 presents estimates for the limits of what one might hope to accomplish in the way of a reduction in the probability of a core melt accident for three typical plants.³ For Oconee-3, for example, it was estimated that introducing a perfect auxiliary feed water system would reduce the probability of a core melt by only a factor of 1.4, while a perfect high pressure injection system would reduce the probability of a core melt by only a factor of 1.1. On the other hand, for the Davis Besse plant which lacked an emergency feed water system, a perfectly reliable auxiliary feed water system would reduce the probability of a core melt by a factor of 21.7, i.e., from 260×10^{-5} to about 12×10^{-5} , an extremely worthwhile improvement.

The SNUPPS Design for a PWR

In the early 1970s, Westinghouse, together with a group of U.S. utilities and Bechtel, evolved a design for a new type of PWR called the Standardized Nuclear Unit Power Plant System (SNUPPS). The prime objective of the design was to reduce cost, but efforts were made also to improve the safety and reliability. Data for a typical SNUPPS design are presented in the second column of Table 1.⁴

The most recent version of the SNUPPS design, and the one in which the greatest attention has been given to the safety problem, was prepared by the CEGB in the U.K. for a plant designated Sizewell B.^{5,6} To meet the stringent requirements posed by the high population density in the vicinity of

the site, the CEGB made an intensive 5-year study of the reactor safety problems and the measures that might be taken to minimize the probability of an accident. This study was exceptionally thorough and included the exploration of a wide range of possibilities and much new design work.

Data for the design evolved are given in the third column of Table 1. The additional steps taken to improve the safety of that power plant include the use of four high pressure safety injection pumps, each having a larger capacity than those normally incorporated on SNUPPS. Four accumulators are provided, each having a capacity sufficient to supply 50 percent of the required flow in an emergency so that any two of the pumps could fail to operate in an emergency without any serious ill effects. Further, low pressure injection pumps are provided to give additional residual heat removal capacity. Two of these are designed to pump borated water to the depressured primary circuit in the event of a Loss of Coolant Accident (LOCA), and two to provide water to the containment spray system. Further, the high pressure injection pumps are designed so that they, too, can perform the low pressure injection operation. In addition, other redundant equipment has been provided in the way of diesels, auxiliary feedwater pumps, and an emergency boration system to provide extra shutdown capability. In addition, a secondary containment vessel has been provided to reduce further the probability of an escape of radioactive material into the environment. The cumulative effects of these various measures designed to make the safety of the plant even better than that of the standard SNUPPS design has been estimated to increase the capital cost about 25 percent.⁷ The PRA for an accident giving severe core damage was 1.2×10^{-6} , while the PRA for such an acci-

dent leading to a large release of activity to the atmosphere was estimated to be 3×10^{-8} .

A firm commitment to construct this plant is expected sometime in the coming year.

The BWR/6 Design

General Electric has evolved the BWR/6 series of boiling water reactors which incorporate a number of important improvements that reduce both the probability of a core melt accident and the probability of the release of significant amounts of radioactivity from the plant in the event any such accident might occur.⁸ Data for a typical BWR/6 design are given in the fourth column of Table 1.⁹ Note that the estimated PRA for an accident that would cause severe damage to the core is 10^{-6} , well below the value of 2.9×10^{-5} used as typical of BWRs in the Rasmussen Report.¹⁰

The principal modification in the design entailed improvements in the natural circulation flow path within the reactor vessel so that there is adequate coolant flow capacity for removing the afterheat from the core by natural thermal convection as long as the water inventory in the reactor vessel is maintained at the proper level.^{8,9} Further, the design provides the capability for rapid depressurization of the primary system so that low pressure as well as high pressure pumps can be employed to maintain the proper water level in the reactor vessel.

Other improvements include new electric-hydraulic fine-motion control drives, increased redundancy, and separated functions. These changes have resulted in a marked reduction in the PRA for an accident that could result

in severe damage to the core. In addition, the containment system has been improved by employing two containment envelopes with a large toroidal water tank placed between them in such a way that gases vented from the inner envelope must bubble through the tank of water. (See Figure 1.) Experiments indicate that the bubbling process should remove between 99 and 99.9 percent of the iodide and particulate fission products in the vented gases. Thus the vast majority of the radioactive fission products that might escape from the core in the event that severe core damage did occur would be retained within the primary containment system. From this it follows that not only is there a very low probability of an accident that would cause severe damage to the core (about 10^{-6}), but, if such an accident were to occur, the probability of the escape of substantial quantities of fission products to the atmosphere would be exceedingly low.

The German Kraftwerk Union Reactors

The Kraftwerk Union in Germany has followed design practices somewhat different from those in the U.S. in developing their line of PWRs and BWRs. These differences stem in part from the greater population density in Germany and thus an even greater concern for reactor safety. Data for one of their largest and most recent PWRs are included in the fifth column of Table 1.¹¹ Note that they employ both a lower power density and a lower heat flux in the reactor core than in the U.S. PWRs, and they provide double containment. Further, their containment vessels are larger, thus making it possible to separate equipment and hence reduce the possibility of common mode failure. The reactor pressure vessels are made larger to reduce the

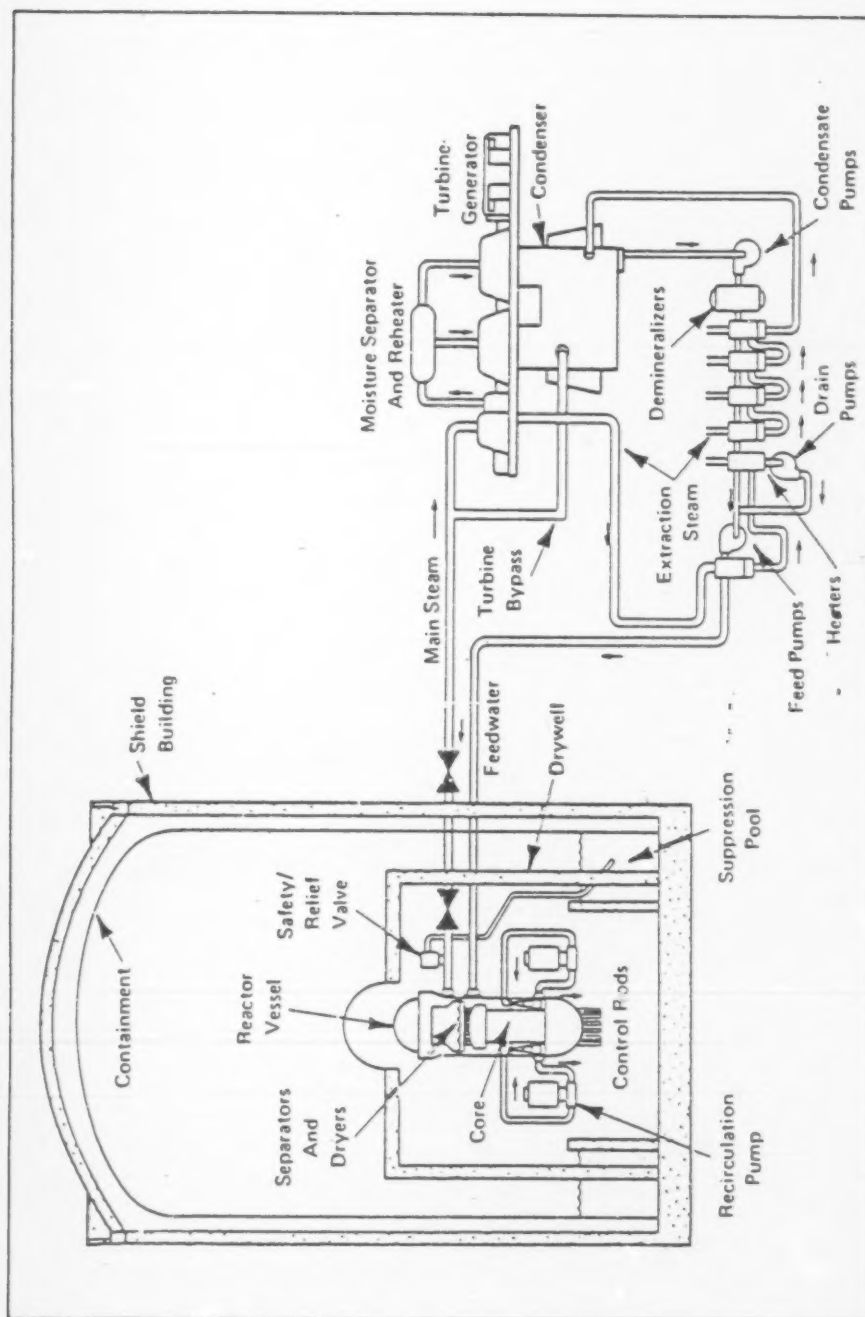


Figure 1. BWR Reference Cycle

fast neutron dose to the steel, and the vessels are made of forged rings to avoid having highly stressed axial weld material sensitive to embrittlement by radiation. In addition, there is more separation of functions in the auxiliary systems and more overdesign in the capacity of the components. There are four independent primary reactor cooling circuits, for example, each with its own circulating pump and steam generator, and each equipped with sufficient capacity in the pumps and tanks of its emergency feedwater system to remove at least half of the afterheat under emergency conditions. This equipment, together with a dedicated set of diesel generators to provide emergency power, is housed in a special shielded building. Both this building and the main reactor containment structure are designed to withstand the impact of an aircraft crash directly on the buildings. These buildings are also designed to withstand both severe earthquakes and attempted sabotage.

Throughout the plant there is extensive use of both redundancy and diversity in the instrumentation, controls, and auxiliaries. These measures are coupled with much physical separation to minimize further the possibilities and/or effects of common mode failures. These features, coupled with the measures outlined above, have led to an estimated PRA of 10^{-5} /year for accidents that might cause serious core damage.

Studies at Karlsruhe and KWU indicate that the probability of activity release from the containment system in the event of severe core damage is very low--of the order of 10^{-4} . Further, in the event of such a release, the amount of activity that would escape would be attenuated by factors of as much as 10^6 because iodine and cesium would remain dissolved in water and aerosols would largely settle or plate out. The magnitude of these

attenuation factors depends on the time interval between initiation of the accident and the point at which leakage from the containment system would begin. Their analyses indicate that at least five days would be required for the pressure in the containment vessel to build up to the point where cracks would develop and leakage could start. These studies also included an investigation of the "China syndrome," and the analyses indicated that a core melt accident could not lead to penetration of more than 4m down into the concrete foundation pad. (German practice is to use a pad thickness of 5 to 7m.)

The rising costs of oil and coal coupled with the large market for relatively low pressure steam for building heating and industrial processes have led Kraftwerk Union AG (KWU) to design a 250 MWt boiling water reactor suited to this purpose.¹² Particular attention has been paid to reactor safety so that the reactor could be located in populated areas. Data for this plant are given in the sixth column of Table 1. Note that the small size and relatively low output of the reactor greatly ease the afterheat removal problem.

Pressurized Heavy Water Reactors

The seventh column in Table 1 summarizes data for a typical pressurized heavy-water reactor.¹³ Although a PRA for severe damage to the reactor core is not available, studies at the Atomic Energy of Canada Ltd. of CANDU reactor systems indicate that such an event is extremely unlikely. For example, in the event of a loss of coolant accident, analyses indicate that the fuel element would heat up rapidly until the zircalloy clad reacted with steam to

form ZrO_2 at which point it would crumble and slump in the channel so that the fuel would reject its heat by contact conductance to the wall of the pressure tube, causing its temperature to rise to $800^\circ C$. Depending on whether or not there was a loss of pressure in the pressure tube, it would then slump or swell until it closed the gas gap between it and the calandria tube, and heat would then flow mainly by contact conductance into the calandria tube and thence into the separate calandria water. When this happened, the temperature of the pressure tube would then drop quickly to less than $200^\circ C$, and heat would be carried off by the heavy water moderator in the calandria moderator region. The UO_2 would not melt, but the volatile fission products would be released into the containment building.^{14 15}

Other factors favoring the heavy-water reactor include the fact that the heavy-water cooling circuit in effect constitutes an emergency core cooling system that is in operation continuously and immediately available for afterheat removal in the event of a failure of the primary circuit pumps. Note also in Table 1 that a typical heavy water reactor has a core power density about 1/10th that of a conventional LWR. From the standpoint of the reactor calandria structure, this high heat capacity ought to more than offset the higher heat flux employed in the fuel element.

Pointing out that CANDU license is predicated on fact that reactor can withstand a LOCA and a loss of energy cooling accident, and a LOCA with openings in the containment (a) CANDU is the only operational power reactor that has two independent shut-down systems, and (b) because of the use of natural fuel, CANDU has a very low built in reactivity. This gives rise to better "controllability" than is the case with LWR's which burn enriched fuel.

NEW SWEDISH CONCEPT FOR AN ULTRASAFE LWR (PIUS)

Although the previous sections indicate that, by clever design and the liberal use of redundant components and systems, it is possible to evolve a system design giving a PRA for a severe core damage accident of the order of 10^{-5} or less, the resulting systems are very complex and the uncertainties that surround the PRA methodology are such that some people question the validity of these numbers. The situation has become so difficult in Sweden that ASEA-ATOM decided in 1979 to go back to first principles and evolve a light water reactor power plant design that would be incontestably safe to operate.^{16,17} Their criteria included not only safety from the standpoint of any conceivable accidents caused by equipment or operator failures, but also safety from sabotage or terrorist attack. Many experts have examined this new concept critically and have been unable to find any series of events, including sabotage and terrorist actions, that appear to have an appreciable probability of causing severe damage to the reactor core. Thus the system is unique and deserves special attention in this paper.

Description

The system envisioned is shown schematically in Figure 2, and data for the reactor are included in the eighth column of Table 1. A large, prestressed, concrete reactor pressure vessel (PCRV) with a cavity diameter of 13 m and a cavity height of about 35 m encloses the entire primary circuit of a PWR. The containment vessel is filled nearly to the top with borated water so that virtually all of the primary circuit is immersed in a pool.

STATIONARY HOT/COLD
INTERFACE IN OPEN
VESSEL (NO BLOCKAGE
POSSIBLE)

17

Feed water

Steam to turbine

MAIN
RECIRCULATION
PUMP

STEAM GENERATO

Core

STATIONARY HOT/COLD
INTERFACE IN OPEN
VESSEL (NO BLOCKAGE
POSSIBLE)

Figure 2. Diagram Showing the PIUS Reactor Concept.

Both the primary circuit and the pool are maintained at a pressure of approximately 9 atm by the pressurizer of the primary circuit which is located in a steam region up at the top of the cavity in the containment vessel. Inasmuch as the pressure drop in the primary circuit is just a few atmospheres, the envelope of the primary circuit separating it from the pool is required to carry only a small pressure differential. However, the hot primary circuit (which runs at $\sim 270^{\circ}\text{C}$) must be covered with a layer of thermal insulation to reduce the heat losses to the pool water (which runs at $\sim 50^{\circ}\text{C}$).

As can be seen in Figure 2, the system has been designed to provide excellent natural thermal convection by placing the reactor core at the base of a hot water column about 30 m high. The density difference between the hot water in the riser above the core and the cold water in the pool is sufficient to give a pressure differential of about half an atmosphere. A large opening is provided between the primary system and the pool water at both the top and the bottom of the core-riser column. Honeycombs are provided in these open regions to inhibit convection currents, while the difference in density between the hot and cold water serves to maintain a stable position for the hot-cold liquid interface in each honeycomb under normal operating conditions.

Control

The system is designed for a constant water flow rate through the primary circuit and a constant reactor outlet temperature over the normal range of power operation. The circulating pumps for the primary circuit provide

sufficient head to overcome the pressure drop through the primary circuit including the core and riser portion. The pump speed is modulated so that the pressure head available for flow through the core and riser is exactly equal to the difference in static head between the hot liquid in the core-riser column and the cold liquid column in the pool. Maintaining a constant water flow and a constant reactor discharge temperature means that the reactor inlet temperature drops as the power increases, and this reduces the static head available for thermal convection. To accommodate this variation, the pump speed is modulated about 3 percent over the range from about 25 to 100 percent power. The static pressure differential between the pool and the core-riser column does not have to be exactly balanced because the height of the honeycomb region is approximately 1.5 m. This serves to accommodate small variations in the difference in head available from the circulating pumps and the static head in the pool. However, in the event of any pump stoppage caused by a loss of electric power or a bearing failure, the flow through the primary circuit pump and steam generator comes to a stop, but thermal convection through the core will begin immediately because of the differential head available for natural thermal convection up through the core and riser and back down through the pool. Of course, as soon as this heavily borated cold pool water enters the core, the nuclear reaction is quenched. Note that no instrumentation and control equipment, valves, or pumps must operate to cope with an emergency. Further, there is sufficient water in the pool so that, if there were to be a concurrent failure in the circuit acting to cool the pool water, the pool water temperature could rise to the point where a pressure relief valve at the top of the containment vessel would begin to release steam, and the afterheat generated from that

point on would be carried off by evaporation of pool water. The capacity of the containment vessel is sufficient to provide cooling water for a week after such an accident before the level of the water would drop to the point where good natural convection cooling induced by boiling in the reactor core would no longer occur.

Protection from Terrorist Attack

The system would be covered by a huge shield plug that could be lifted out of the way only with the overhead crane provided in the building. After this shield plug had been put in place over the reactor, the crane would be disassembled so that no terrorist could employ it to remove the shield plug and sabotage the reactor. At least two days would be required to reassemble the crane, and in that time it should be possible to cope with terrorists. An examination of this and other conceivable scenarios by several dozen experts has failed to yield any credible course of events that could lead to serious damage to the core and hence the potential for a substantial release of fission products. As a consequence, there appears to be no plausible sequence of events that could result in either serious damage to the core or a serious release of radioactivity to the atmosphere.

Crucial Problem Areas

The principal concerns expressed by those who have reviewed the Swedish concept have been with respect to the stability and control of the interfaces between the cool pool water and the hot primary circuit (particularly

under transient conditions), the capital cost of the large prestressed concrete pressure vessel, and the difficulties of maintenance through the relatively small opening at the top of the vessel. The capital cost and maintenance questions are addressed in the last section of this report where different types of reactors are compared, but the most frequently expressed objection, i.e., possible difficulties in the control of the hot-cold liquid interfaces is peculiar to this reactor and merits a few words here. The basic design study in Sweden included the development of a digital computer model of the system and the investigation of a fairly wide variety of severe transient conditions; the results indicated that the system was stable and well-behaved under every condition considered. However, the various factors affecting the position of the hot-cold interfaces are surprisingly complex for this otherwise simple system, so much so that the computer model is quite involved and its validity a bit difficult to assess. In view of this, the writer made an independent effort to evolve a rationale for the extent to which the hot-cold liquid interfaces would be perturbed by transients. Space does not permit a delineation of either the Swedish or the writer's approach, but suffice it to say that both indicate that the variations in the level of the free liquid surfaces in the course of normal transient conditions should not cause the highly borated pool water to enter the primary circuit.

THE HIGH TEMPERATURE GAS-COOLED REACTOR (HTGR)

The experience gained in the operation of the Dragon Reactor in England, the Peach Bottom Reactor of Philadelphia Electric, and the AVR in

Jülich, Germany (all three were experimental reactors), and the Fort St. Vrain commercial reactor near Denver provide a good basis for assessing the operability, maintainability, safety, and costs of high temperature gas-cooled reactors. To assess their potential, it seemed best to include in the two right-hand columns of Table 1 two post-TMI designs, one a General Atomic design for a 2240 MWt reactor,^{18,19,20} and the other a German modular design for a 250 MWt, 100 MWe modular unit that could be employed in groups, e.g., four would give a 1000 MWt, 400 MWe plant.²¹

The two HTGR designs of Table 1 have a number of inherent characteristics that are advantageous from the safety standpoint. (Note in Table 1 that the PRA for severe core damage to the 2240 MWt reactor was estimated to be quite low, i.e., 40×10^{-6} .) Probably most important is the relatively low power density of HTGRs--between 5 and 10 percent that of a conventional PWR. Further, inasmuch as the fuel is dispersed throughout the moderator, the heat capacity closely associated with the fuel is over 100 times that for an LWR. Thus, in the event of any accident that interrupts the flow of coolant to the core, the elapsed time between the cessation of coolant flow and severe damage to the core (if no automatic or operator action is taken) is of the order of 10 h in an HTGR, rather than only 45 s for a PWR, the difference being a factor of 1000. One factor contributing to this is that lateral heat conduction through the graphite blocks in the HTGR core is sufficient to remove a substantial fraction of the afterheat from the core and carry it out through the reflector, from which the heat is radiated to the water-cooled steel liner of the reinforced concrete pressure vessel. In fact, in the small 250 MWt size chosen for the modular plant (the righthand column of Table 1), heat conduction via this route is sufficiently good even

in a pebble-bed type of reactor so that in the event of a loss of coolant accident the fuel temperature will not reach a high enough level to cause any major release of fission products. Tests in the AVR reactor at Jülich have demonstrated this capability. Thus, in the 250 MWt HTGR, the PRA for an accident that could cause severe damage to the reactor core should be extremely small.

In the prestressed concrete pressure vessel layout contemplated for the 2240 MWt HTGR of Table 1, cost considerations have favored placing the steam generators at approximately the same level as the reactor, hence natural thermal convection is not sufficient to remove the afterheat from the core. However, in the 250 MWt HTGR modular plant design a tall cylindrical steel vessel has been employed with the steam generator placed above the reactor so that natural thermal convection of the helium coolant is adequate for afterheat removal.

The core diameter for the modular design was limited to around 3 m so that it is possible to get adequate reactivity control just with control rods in the reflector, i.e., with none in the core. This avoids the possibility that a control rod might become ineffective because it would fail to penetrate an unfavorable fuel ball configuration in the pebble bed.

Another advantage of the small size of the modular design is that a unit could be shop-fabricated and shipped to the site. The 250 MWt, (100 MWe) unit, for example, has the reactor and steam generator installed in a cylindrical pressure vessel 6.2 m in diameter and 35 m high. Shop fabrication should lead to major reductions in cost and construction time as well as yield a higher quality in the welds. Note that U.S. experience in conventional coal-fired steam generators indicates that the incidence of ser-

vice failures in shop welds is only a few percent of that in field welds.²² These advantages of the modular design from the cost and maintenance standpoints may be offset by the increase in the amount of instrumentation and control equipment needed because each of the modular units will require a full set of such equipment, and some additional equipment will be needed to operate a multiplicity of units in parallel. The data available to the writer are not sufficient to provide a basis for firm conclusions with respect to these effects.

SUMMARY OF CHARACTERISTICS OF ADVANCED REACTORS

Any effort to compare post-TMI upgraded versions of established nuclear plant designs with designs for advanced types of nuclear plants is inherently vulnerable to criticism. As a consequence, for the purposes of this paper the characteristics and figures of merit chosen for comparison are those that seemed less subject to debate. The information available was organized in Table 4, and appears to be sufficiently comprehensive to provide a good basis for comparison of the various reactor systems.

Safety

The most significant parameter from the safety standpoint appears to be the PRA for severe damage to the reactor core. The NRC, the ACRS, and the AIF have all taken the position that the design goal for severe damage to the core should be 10^{-4} or less per reactor operating year for new reactors. This low value coupled with a good containment system should give a

TABLE 4. SUMMARY OF THE CHARACTERISTICS OF THE PRINCIPAL TYPES
OF POST-TMI NUCLEAR PLANT DESIGNS

	PWR	BWR	CANDU	PIUS	Large HTGR	Modular HTGR
Safety as indicated by PRA for severe core damage	5 x 10 ⁻⁶	10 ⁻⁶		<<10 ⁻⁶	40 x 10 ⁻⁶	
Vulnerable to sabotage or terrorism?	Yes	Yes	Yes	No	Yes	Yes
Availability, Worldwide average, %	58	56	79			
Best 5y period of best plant, %	86	80	87* **			
Cost relative to PWR	1.0	1.0	1.34	?	1.2?	?
Fuel cycle development required	None	None	None	None	Some reprocessing	Some reprocessing
Ease of regulation	Complex	Complex	Complex	Simplified	Complex	Complex
Construction	Complex	Complex	Complex	?	Complex	Good
Operability	Complex	Complex	Complex	?	Complex	?
Radiation dose levels affecting maintenance	5 to 40 r/hr @ steam gen.	1 r/hr @ turb. during op; 3 mr 1d after shutdown			10 mr/h @ steam gen.	Probably low

*This value is for the best capacity factor because a corresponding value for the availability could not be found. The availability would certainly be higher.

**If cost of LWR includes fully loaded core (enriched fuel) then the cost of CANDU, fully loaded w/HW is compatible not 34% higher as shown.

PRA for a major fission product release to the atmosphere of 10^{-6} or less, and thus reduce the hazard to the public to a trivial level. An examination of Table 4 indicates that all of the types considered here appear to meet this criterion, and in several cases the estimated PRAs are even lower by factors of 20 to 100. This is a remarkable and highly encouraging development. However, it is not clear whether the public, and more particularly leaders in the antinuclear movement, will accept these estimates, in part because the problems are complex and the rationale subtle and abstruse and in part because there are large uncertainty bands in the PRAs. In this respect the PIUS concept has a major advantage in that, from the safety standpoint, the system is sufficiently simple so that the rationale for its remarkably good safety characteristics can be comprehended by anyone who will make an effort to understand it. It may be that this relative simplicity is so important, and the consequent reduction in licensing and construction costs and time are sufficiently great to make this (or a similar approach) the preferred option.

The PIUS system has another major advantage; the probability that an act of sabotage or terrorism may cause severe damage to a nuclear plant and even endanger the public is a matter of some concern. All of the plant designs considered here appear to be potentially vulnerable to sabotage or acts of terrorism except for the PIUS design; this offers the highly important possibility of building a plant that would be invulnerable to any malevolent behavior.

The novel approach taken in the PIUS design has stimulated efforts to evolve other designs having similar highly attractive advantages. Unfortunately, none of these concepts is supported by the degree of detailed design

that has been carried out for the PIUS concept, but the prospects of getting other design concepts with similar advantages appear to be promising. One such concept for a steam-cooled fast breeder reactor, for example, has some especially unusual and attractive features.²³

Reliability, Maintainability, and Availability

Questions of reliability, maintainability, and availability are difficult to evaluate from paper designs, but in cases where a substantial number of plants have been in operation for quite a number of years, it is possible to cite statistical results for availability. Table 4 includes such a set of data.²⁴ The differences appear to be significant, but much of the loss of availability of LWRs in the U.S. is undoubtedly chargeable to licensing and safety requirements that have imposed many outages in recent years for retrofitting of additional safety-related equipment. Thus it is also significant to include data for the best five-year average for the best plant of each type to indicate what might be obtained.

The availability of a power plant depends largely on the reliability and maintainability of components. It is significant that the heavy water reactor with its many calandria tube connections is the most complex of the reactors of Table 4, yet it has had the highest availability. There is no basic difference in the corrosion problems or in the basic design of the pumps and steam generators between it and the other LWRs except for much more stringent requirements on leak tightness imposed by the high cost of leakage losses from heavy water systems. Paradoxically, this tough requirement has evidently demanded such high quality component design and develop-

ment work that the resulting system has demonstrated a higher availability than the LWRs with their less demanding design requirements. Part of the difference undoubtedly stems from the on-stream refueling capability of the CANDU reactors, but the record indicates that they must also have a higher reliability. It appears to the writer that the excellent demonstrated availability of the CANDU reactors is more a tribute to the designers than an attribute of the reactor type. It should be added that most designers of equipment such as pumps and steam generators follow empirical procedures that all too often overlook differences between conditions that will prevail in a new application and those that characterized the former applications from which the empirical design rules were derived. Now that extensive experience has been gained with LWRs, much improved design bases for steam generators and other components have been evolved. There is no reason why these components should not have a life of 40 y with a very high reliability. This, in turn, should greatly increase the plant availability while at the same time drastically reducing maintenance requirements.

The last item in Table 4 gives data on radiation dose levels after shutdown to indicate something of the difficulties of maintenance operations. Inasmuch as NRC standards on occupational doses limit the dose to personnel to 1.25 r per quarter, it is generally impractical to carry out contact maintenance in areas where the dose level exceeds about 100 mr/h. Thus the dose levels in the vicinity of primary system pumps, steam generators, and piping as given in Table 4 are generally too high in LWRs to permit contact maintenance.^{25 26} While the dose levels in the vicinity of the recirculating pumps in the BWR are about the same as for a PWR primary system pump, the dose levels at the turbine are surprisingly low—only around 2

to 3 mr/h a day or two after shutdown.²⁷ This result is particularly remarkable in view of the fact that, when BWRs were first proposed, it was feared that dose levels at the turbine would be quite large after shutdown. Apparently the principal mechanism responsible for long-lived activity in the water reactor circuits is a trace amount of corrosion in the water system and conveyance of the corrosion products in liquid water to points outside the reactor. These radioactive corrosion products evidently are not transported by steam. The most important of these radioactive corrosion products transported by water are cobalt 58 and 60 coming from the trace amounts of cobalt in the stainless steel in the reactor core and stellite bearings and valve seats elsewhere in the system. There seems to be little transport of radioactive materials in gas streams. Note in Table 4 the surprisingly low dose level in the vicinity of the steam generators of the HTGR; this is sufficiently low to permit contact maintenance.

Another indication of the problems posed by transport of radioactive material outside the reactor core is given by Figure 3 which shows the annual exposure to operating personnel in man rem per unit reactor for BWRs, PWRs, and the Fort St. Vrain HTGR.²⁸ Note that the annual exposure of plant personnel runs less than one-tenth as great for the HTGR relative to the PWR and BWR reactors.

The above considerations are particularly important in attempting to assess the reliability, maintainability, and availability of a new concept such as the PIUS design, or a reactor type such as the HTGR with which there has been relatively little experience. For the PIUS concept the problems of maintaining the equipment in the primary circuit inside the containment vessel of Figure 2 are certainly formidable. However, the amount of activated

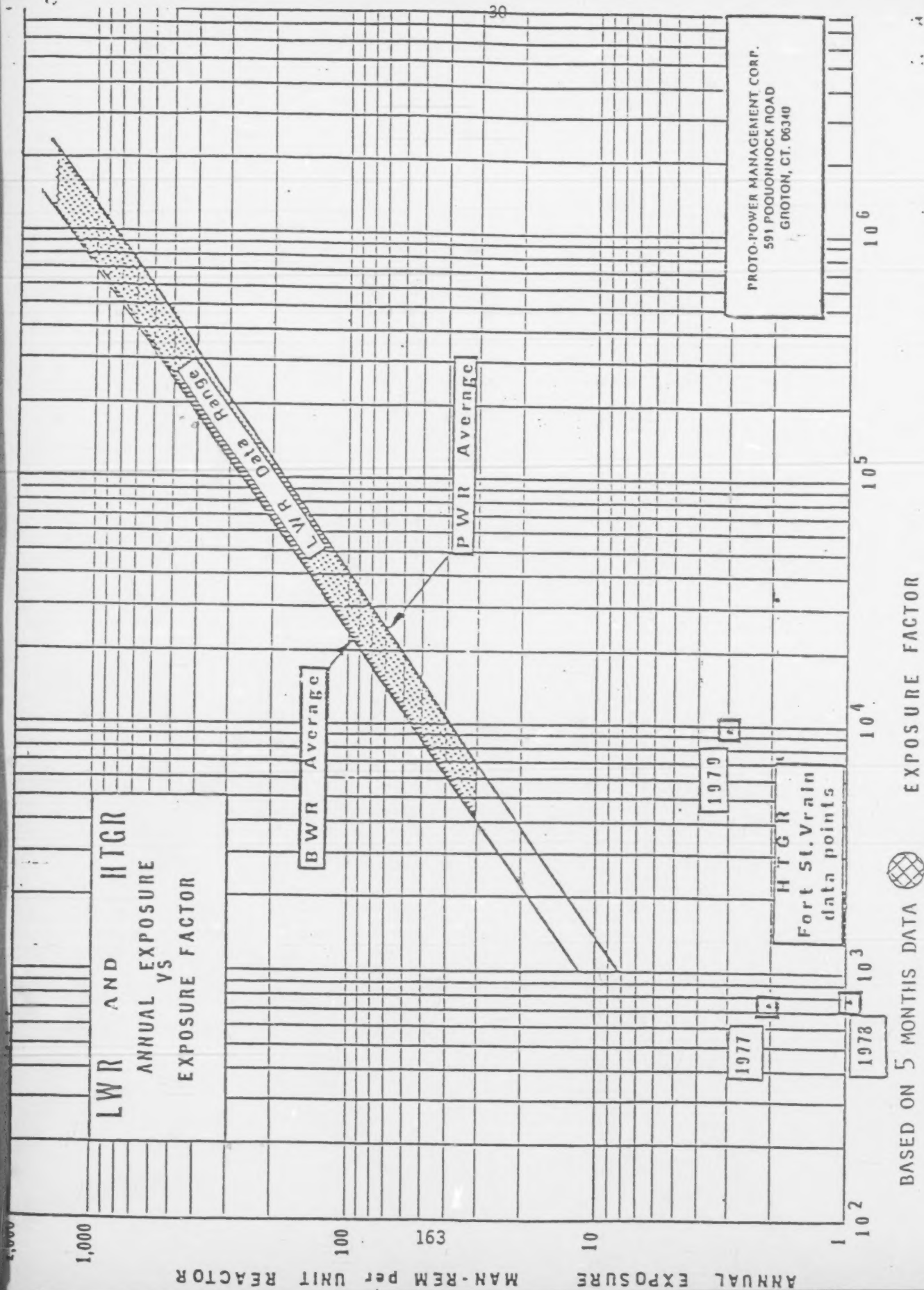


Figure 3. Comparison of Radiation Exposure Rates for BWR, PWR, and HTGR Units.²³

structural material transported from the reactor core out to the steam generator and other portions of the primary circuit in PWRs is such that dose levels of 1 r/h to 40 r/h commonly prevail in the vicinity of the steam generators and pumps. This is much too high for contact maintenance. Awkward temporary shielding is required, and the time required to change out these components is commonly many months.²⁶ For the plant design of Figure 2 it is clear that remote handling equipment must be employed for all components in the primary circuit. This means that both the layout of the primary circuit and the detail design of its components should be developed concurrently with the design of the remote handling equipment projected for the maintenance operations. If this were done properly, it seems possible that the overall costs for maintenance of a PIUS plant might turn out to be less than has been the case for conventional PWRs, particularly if this should make it possible to carry out these maintenance operations in much less time than has been the case for PWRs. Note that one major cost factor associated with maintenance is lack of availability; shutting down a nuclear power plant costs on the order of \$500,000 per day. In short, good design work at the outset should make it possible to obtain much lower maintenance costs than ad hoc improvisation after trouble develops in an operating plant. This thesis applies to any type of reactor, not just the PIUS concept, and the validity of this thesis is indicated by the outstanding availability record achieved with the CANDU plants.

Capital Costs

Capital costs have escalated so rapidly and surprisingly that it is difficult to assign values for the various types of plant. In general it

appears that difference in the costs of BWR and PWR plants is much less than the variation in the costs of nuclear plants built at different sites and by different contractors. An indication of the cost of CANDU reactors relative to conventional PWRs is given by a study made at the Argonne National Laboratory. This yielded the estimate that, mainly because of the heavy water inventory, the capital cost of the CANDU type of reactor would be approximately 34 percent higher than for a conventional PWR.²⁹ However, this higher capital cost would be partly offset by a lower fuel cost.

In turning to the question of the capital cost of the PIUS concept, as indicated near the beginning of the report the difficulties in assessing the capital cost of new types of nuclear plants are so great that a meaningful comparison between various types of reactor is difficult indeed. However, Table 5 presents data for the most significant of the major cost parameters, namely the weight of steel and the volume of concrete required for some typical cases. Note that the values for these parameters for the PIUS concept are below those for the two pressurized water reactor cases (Seabrook and WPPSS) as well as those for the HTGR study. Note, too, the large dif

TABLE 5. COMPARISON OF MATERIAL REQUIREMENTS FOR TYPICAL REACTORS

	1200 MWe* Seabrook	1200 MWe WPPSS	1000 MWe HTGR Study**	2 x 2500 PIUS study (concrete vessels only)
Steel, tonees	34,000	53,000	25,300	19,200
Concrete, yd ³	162,500	252,000	134,000	89,000

*J. H. Crowley testimony before WPPSS.

**ERDA-109, "1975 Technology"

ference between the Seabrook and WPPSS plants, both of which are pressurized water reactors. One factor leading to differences between these various cases is probably the extent to which they include structures outside the reactor containment envelope, particularly such items as the turbine bay and the condenser cooling water channel. Inasmuch as the extra costs of licensing and quality control are often overriding factors in the overall cost of the power plant, one would think that the excellent inherent safety characteristics of the PIUS concept might lead to major reductions in the cost of the instrumentation and control equipment, licensing, and the quality control requirements for the nonnuclear portion of the plant. It would probably be necessary to build and operate a number of these plants before the possibilities could be evaluated.

Inasmuch as only one large HTGR has been built in the U.S., convincing capital cost data are not available for the HTGR. It seems generally agreed that HTGRs do entail higher capital costs, and one large utility has estimated that an HTGR would cost them about 20 percent more than a PWR. This value has been used in Table 4 with a question mark to indicate a substantial uncertainty.

Fuel Development Status

All of the water reactors employ UO_2 as the fuel. The various operations for fuel preparation, processing, fabrication, and reprocessing have been well worked out. Although there are some uncertainties associated with the fabrication of highly radioactive reprocessed fuel, these uncertainties are such that they could be answered only by large scale operations. The

preparation and fabrication of the coated particle fuel for the HTGRs is well developed and the soundness of the resulting fuel has been well demonstrated in the Dragon, AVR, Peach Bottom, and Fort St. Vrain Reactors. However, for the HTGR, the cost of fuel reprocessing and the use of thorium as a fertile material for breeding ^{233}U , together with the reprocessing and re-fabrication of such fuels, will entail questions that require substantial further development work before a large scale operation could be undertaken. The HTGR has also been proposed as a heat source for high temperature process heat (950 °C), and this would require some further development of the coated particle fuel.

Regulation, Constructability, and Operability

The problems of government regulation of nuclear power plants, particularly from the standpoint of safety, depend in large measure on the complexity of the plant. Ample experience is now at hand to show that the complexities of PWR, BWR, and CANDU plants lead to complex problems in government licensing and regulation. The only reactor type in Table 4 that appears to offer the possibility of greatly easing regulation problems is the PIUS design. Similarly, the problems of constructing and operating various types of plants seem to be complex for all of the types considered here, although the PIUS plant and the modular HTGR appear to be less complex. For these latter two cases of the prospects for shop fabrication of major components, relatively much less construction work in the field, and inherently good removal of afterheat may make the quality control requirements for the balance of the plant no greater than those for conventional coal-fired plants. This

should greatly ease the problems of construction and operation as well as reduce costs.

QUALITATIVE PROS AND CONS OF ALTERNATIVE NUCLEAR REACTORS

Light Water Reactors (LWRs)

Pros

- The existing U.S. nuclear energy system is based almost exclusively on LWR (both PWR and BWR) technology. Extensive experience has advanced the learning curve. The various institutions involved in the nuclear enterprise are almost exclusively geared to LWR technology, and the U.S. investment in LWRs is of the order of 100 billion dollars.
- LWRs are the most readily licensable reactor in this country. No new research "frontiers" need be broached to obtain licensing.
- The capital costs of constructing LWRs appear to be less than current-generation alternative reactors.
- Significant safety improvements can be made in next-generation LWRs, witness the Sizewell B and KWU design examples. These reactor designs more than meet the quantifiable safety goals now being set forth by the NRC.

Cons

- Though the capital costs of LWRs are less than for current-generation alternative reactors, they are still high compared to the coal alterna-

tive. The uncertainties facing the industry today could lead to greater cost escalation, further souring utility and financial market perspectives of the nuclear alternative. The addition of new safety measures could seriously increase costs.

- A significant segment of the public remains skeptical of the operating safety of existing LWRs, and is not likely to be swayed by the incorporation of incremental safety improvements.

Heavy Water Reactors (HWRs)

Pros

- HWRs have been used successfully in Canada for many years. They have demonstrated levels of availability considerably in excess of those associated with LWRs.
- HWRs are more fuel-efficient than LWRs.
- HWRs (CANDU) incorporate inherent safety features notably:
 - horizontal pressure tubes
 - two independent shut down systems
 - particularly low built-in reactivity, hence good controllability
 - a large heat sink-in case of a core melt the calandria heavy water moderator would carry heat away.

Cons

- Considerable time and money would be needed to ensure the licensability of the HWR in the U.S. because the HWR differs in many respects from the LWR.

- It appears that the capital costs associated with building a licensable HWR could be as much as a third more than LWR capital costs. Heavy water production capacity would also be expensive.
- There is no U.S. vendor geared to producing HWRs. Canadian experience with HWRs is centralized in one province and one utility. Extensive technology transfer to the U.S. would pose difficulties.
- Though probably not serious, some new environmental issues would have to be addressed. Because of lower thermal efficiencies than LWRs, approximately 20 percent more waste heat is produced. Tritium releases would also require investigation. A thorough PRA has not yet been completed for an HWR, hence there may be some low-probability, high-consequence accidents that will pose licensing problems.

High-Temperature Gas-Cooled Reactors (HTGRs)

Pros

- HTGRs have several inherent safety advantages over LWRs, the foremost being a lower power density and a larger core heat capacity.
- The higher operating temperature of HTGRs gives a greater operating efficiency, hence less waste heat and less nuclear waste products per unit of output. The transport of radioactive material out of the reactor is less than in LWRs so that the radiation dose to maintenance personnel is much reduced.
- HTGR development has been underway since the 1950s and the operating commercial reactor at Fort St. Vrain (330 MWe) is providing valuable

experience. An HTGR vendor (The General Atomic Company) already exists in the U.S.

Cons

- HTGRs are in the prototype development stage. Prior to the licensing of large HTGRs a number of problems would require further investigation.
- It appears that the capital costs associated with building a licensable HTGR might be 20 percent higher than a comparable LWR.

Process Inherent Ultimately Safe Reactors (PIUS)

Pros

- The PIUS reactor goes back to first principles to ensure that no severe core damage is possible. It is the only reactor type for which the safety features are so transparent that it appears possible to convince a skeptical public that there is no danger of radioactive releases from component failures, operator error, or from saboteurs or terrorists.
- The PIUS reactor could be exported with confidence to countries where the expertise and capabilities to handle a more demanding nuclear reactor might be questionable.
- The PIUS reactor incorporates much LWR technology; yet because of its inherently safe features, it should not be subject to the increasing cost escalations affecting LWRs by continuing requirements to add more

safety features to the reactor. Urban siting may even be feasible, with its consequent economic savings.

Cons

- The PIUS reactor concept has not yet been carried beyond the design stage. To develop and prove the technology would take a long-term commitment from either a reactor vendor or a government. Whether the PIUS reactor will make sense in the political and social environment of the late 1990s and early 2000s is anyone's guess.
- Without testing and operating experience, there are large cost uncertainties. No assurances can be given, therefore, that the PIUS will be an economical power producer.

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NUCLEAR POWERPLANT SIZE
DEPENDENT CHARACTERISTICS

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1.0 INTRODUCTION

Nuclear power reactors for the generation of electricity have been built in sizes ranging from around ten to 1300 MWe. Plants currently being built range from around 900 to 1300 MWe. For any type of machine or structure, the design will eventually evolve to a size which is in some sense optimum for the mission requirements and economic constraints. As the requirements and cost constraints change, the optimum size may also change, as witnessed by the recent downsizing of automobiles, and coal fired generating stations.

It is our purpose to explore the role of reactor size in the current nuclear industry problems, and to consider potential advantages to be gained by reducing the size of future nuclear power plants. Historical data on the performance of small and large domestic nuclear power plants is considered, together with feedback from power plant designers, operators, regulators and concerned members of the public. Design features which could be incorporated into a new generation of smaller plants are surveyed, as are the activities in small reactor design by other supplier countries.

1.1 Background

The first nuclear power plants were built as "demonstration plants" to show the potential of nuclear power for electricity generation. They ranged in size up to about 200 MWe. At the time they were built, these

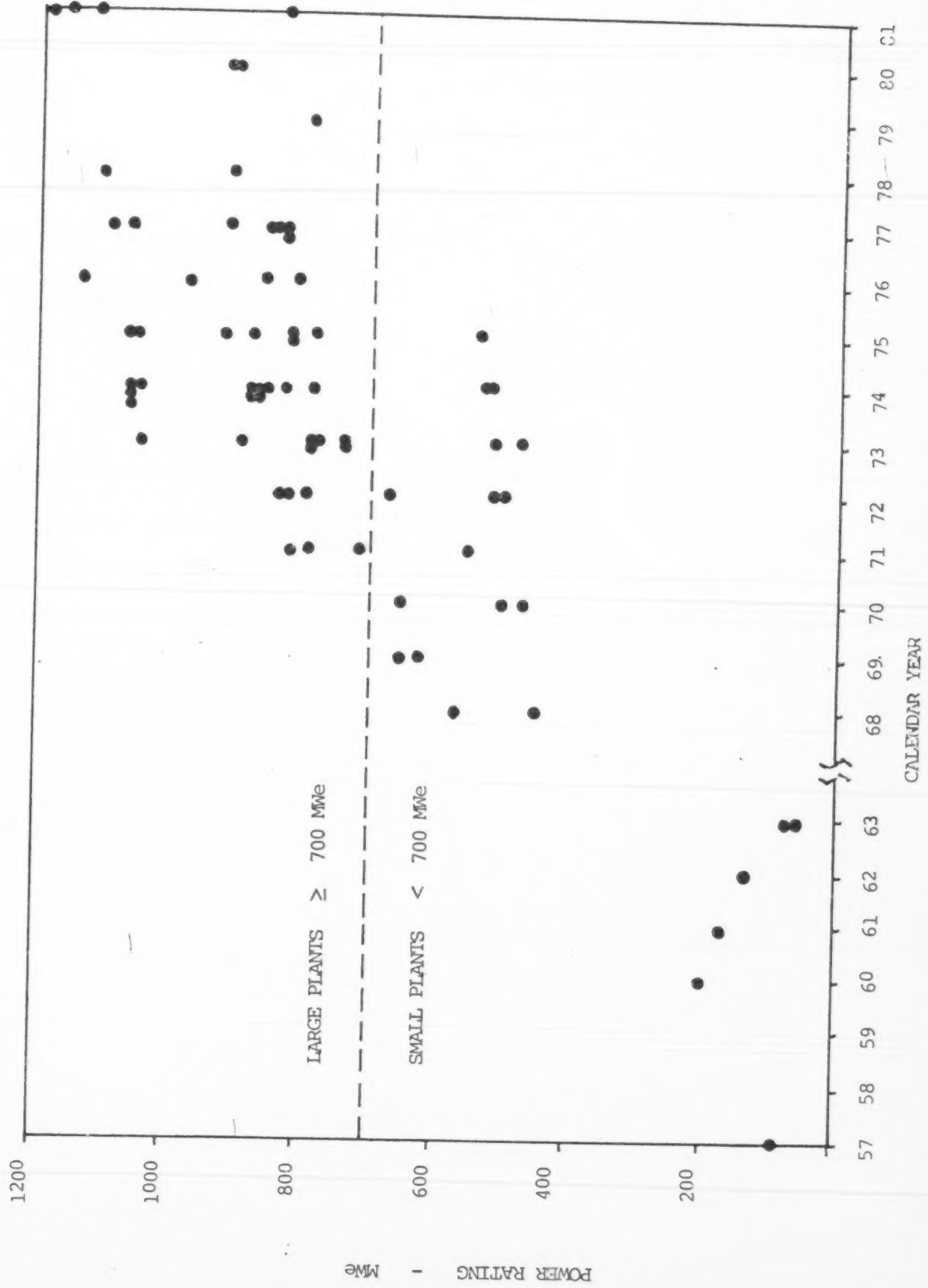
plants were not competitive with fossil generation. The first nuclear plant which was sold to a utility at a cost competitive with coal fired generation is widely believed to be the Oyster Creek plant which began commercial operation in 1969. One reason for the competitiveness of nuclear power was the economy of scale achieved by building larger plants. Oyster Creek at 650 MWe was thought to be the size necessary for a nuclear plant to be competitive with coal fired generation. Even larger plants were built to make nuclear competitive in all areas of the country. For the economic conditions of the 1970's, there was general agreement that the economy of scale clearly favored the large units. Figure 1 shows the size (Net MWe) of Light Water Reactor (LWR) units which began commercial operation in each year through 1981. The increase in unit size with time up to the present 1000 MWe size range is evident. For purposes of later discussion, it is convenient to distinguish three size classes:

- large plants, greater than 700 MWe
- small plants in the range 200 to 700 MWe
- demonstration plants (less than about 200 MWe)

Note that a number of small units were built and brought on line during the 1968-1975 time period.

The rapid scale-up in size of nuclear units was driven by the economic conditions of the period. The degree to which the large reactor designs were scale-ups of earlier smaller reactors differed from vendor to vendor. However, in all cases the scale-up, such as it was, occurred very rapidly as is evidenced by the data shown in Figure 1.

FIGURE 1 POWER RATING OF NEW COMMERCIAL REACTORS



The first scale-up occurred in going from the 200 MWe or less demonstration plants to the first truly commercial 600 MWe units. This was a significant increase in plant size and was followed by a further scale-up to the 1000 MWe range over about a five year period. Three of the four NSSS vendors built at least one plant in the 600 MWe range. Westinghouse plants in this power range featured two coolant loops and two steam generators. The larger units had approximately the same size loop components, but featured three or four coolant loops. General Electric built five BWRs in the 600 MWe range and ten in the 800 MWe range. Combustion Engineering built the Ft. Calhoun 500 MWe reactor. Modern B&W plants are all greater than 700 MWe. An early B&W plant, Indian Point 1, is around 200 MWe and more closely resembles the Shippingport reactor, than the larger modern plants. Like Shippingport, which was developed in the Naval Reactors program, Indian Point 1 has four coolant loops with four small steam generators, and is the only other plant with inlet nozzles near the bottom of the vessel. Modern B&W plants have two much larger steam generators of a different design and a different coolant loop configuration. They are not a scale-up of the Indian Point 1 design and, hence, in the case of B&W, there was no scale-up of a small reactor design. The original design is greater than 700 MWe.

1.2 Factors Affecting Power Plant Size

During the sixties, it was widely believed that power plant construction costs scaled according to the "Six-Tenth Power Law".⁽¹⁾

That is,

$$\text{Cost} \propto \{ \text{Capacity} \}^P$$

where $P = 0.6$. This belief led to economic pressures to build nuclear plants of the same size as the largest coal-fired units. For these large plants, nuclear was competitive despite its higher capital costs by virtue of its much lower fuel cost.

The maximum size of nuclear plants was limited by the ability to fabricate the large components required. Turbine/generator sets for large fossil plants had been built in sizes up to the 1000 MWe range. Facilities were developed in the industry to fabricate nuclear components for plants in this size range. Current regulations limit the maximum size of nuclear units to 3800 MWt, which corresponds to about 1300 MWe. Therefore, the size range for the present generation of LWRs is between 500 and 1300 MWe.

1.3 Present Industry Climate

Because of the large uncertainties associated with the cost and financial risk of nuclear plants and very slow load growth, there have been no domestic orders for nuclear power plants for almost four years. Cancellations and deferrals of units on order and under construction are continuing at a steady pace. There are a number of industry people who believe that there will be no domestic nuclear plant orders in this decade. The NSSS vendors are in the process of shifting their emphasis from hardware to service operations. While key NSSS vendor management

have recently stated their intention to stay in the nuclear business, and to be ready for a resurgence of orders ⁽²⁾, it does not appear likely that significant investments will be made to develop new LWR designs. The present market for new plants of any kind is extremely limited.

Some limited activity, however, is going on in the development area. Westinghouse is presently working on an advanced PWR design (APWR) in cooperation with a Japanese firm (Mitsubishi). This large PWR design is aimed at achieving very high reliability and fuel efficiency. General Electric is also participating in the joint development of an advanced BWR (ABWR) with several Japanese utilities and other firms including Hitachi Ltd. and Toshiba Corp. This large reactor effort is directed at simplification and evolutionary improvements. B&W and UE&C are doing some conceptual design and costing studies, under DOE sponsorship, for a 400 MWe Consolidated Nuclear Steam System (CNSS) ⁽³⁾. This design has all components, with the exception of the pressurizer, located within the reactor vessel. General Atomics is currently working on a small HTGR design for electricity generation. ⁽⁴⁾ Ft. Saint Vrain at 330 MWe falls in the small reactor category. HTGR plant concepts with inherently safe features, such as passive decay heat removal, are also being explored. ⁽⁵⁾

In summary, U.S. NSSS vendors have an underlying interest in, and are spending a small amount of effort studying, new or improved reactor designs. Some of these efforts are for small reactors. However, given the current state of the market for reactors of any type, it is virtually

unthinkable that any vendor will commit to a new design until the climate in the nuclear industry improves dramatically. There is also some question as to the amount of investment vendors will be willing to make, in light of past experience in this market. Architect/Engineers working on a time and materials basis have profited in the power plant market, while NSSS vendors working on fixed price agreements have had to rely on selling service rather than hardware for their profits.

2.0 SIZE DEPENDENT NUCLEAR POWER PLANT CHARACTERISTICS

While the scale-up of nuclear power plant size was driven by economies of scale, there are also technical and safety characteristics which change as a function of reactor size. These include such characteristics as opportunities for shop fabrication and standardization, siting and licensing considerations, reliability, ease of maintainance, etc. The following sections discuss the influence of plant size on technical, safety and cost factors.

2.1 Technical Factor

There are no major additional components required for large nuclear plants that are not required for small plants. Differences are due to size and simplicity of components themselves, rather than the need for additional components in large plants.

2.1.1 Construction Techniques

A review of recent nuclear power plant construction costs shows that certain costs and delivery schedules have been held nearly constant, while other costs have drastically increased. It is interesting to identify the areas of concern, and to provide some speculative reasons for such conditions. The benefits that may be achieved in these areas by development of smaller size nuclear power plants are considered.

It is well known that there are plants currently under construction that were planned to cost approximately \$400 million and are currently approaching four billion or ten times the original estimated cost for

construction. Some of the increased costs can be attributed to adding more quantities of materials, but such increases are not the full reason for the added costs. Budwani⁽⁶⁾ has estimated the percentage increase in materials for a two unit BWR scheduled to begin operation in 1983-84, compared to a similar plant which began operation in 1974. The percentage increases range from a low of 8.3% for small diameter piping to a high of 96.8% for steel. Crowley and Griffith⁽⁷⁾ have found similar trends in construction material requirements for PWRs. Ranges are given for material requirement increases based on data from eight PWRs. While some of the material requirements show significant increases, the increases are less than overall cost increase percentages.

It is worth considering the types of initial contracts that launched the project. Specifically considered are the type of contracts drafted between the Architect Engineer (A/E) and Construction Manager or utility. Where a design is clearly presented, it is usually easier to obtain fixed price contracts, while cost plus contracts are usually used when a design is not as clearly defined. The advantages of having a stabilized design can be seen when the cost of the nuclear steam supply system and turbine generator set are compared with the total costs of nuclear plants either recently completed or currently under construction. While the cost of a nuclear steam supply system may increase due to scope changes, it will probably never double in costs from the initial contract. Likewise, a turbine generator set is unlikely to increase in cost by more than twenty percent in spite of escalation clauses. Therefore, the main equipment items that make up a nuclear plant are not responsible for

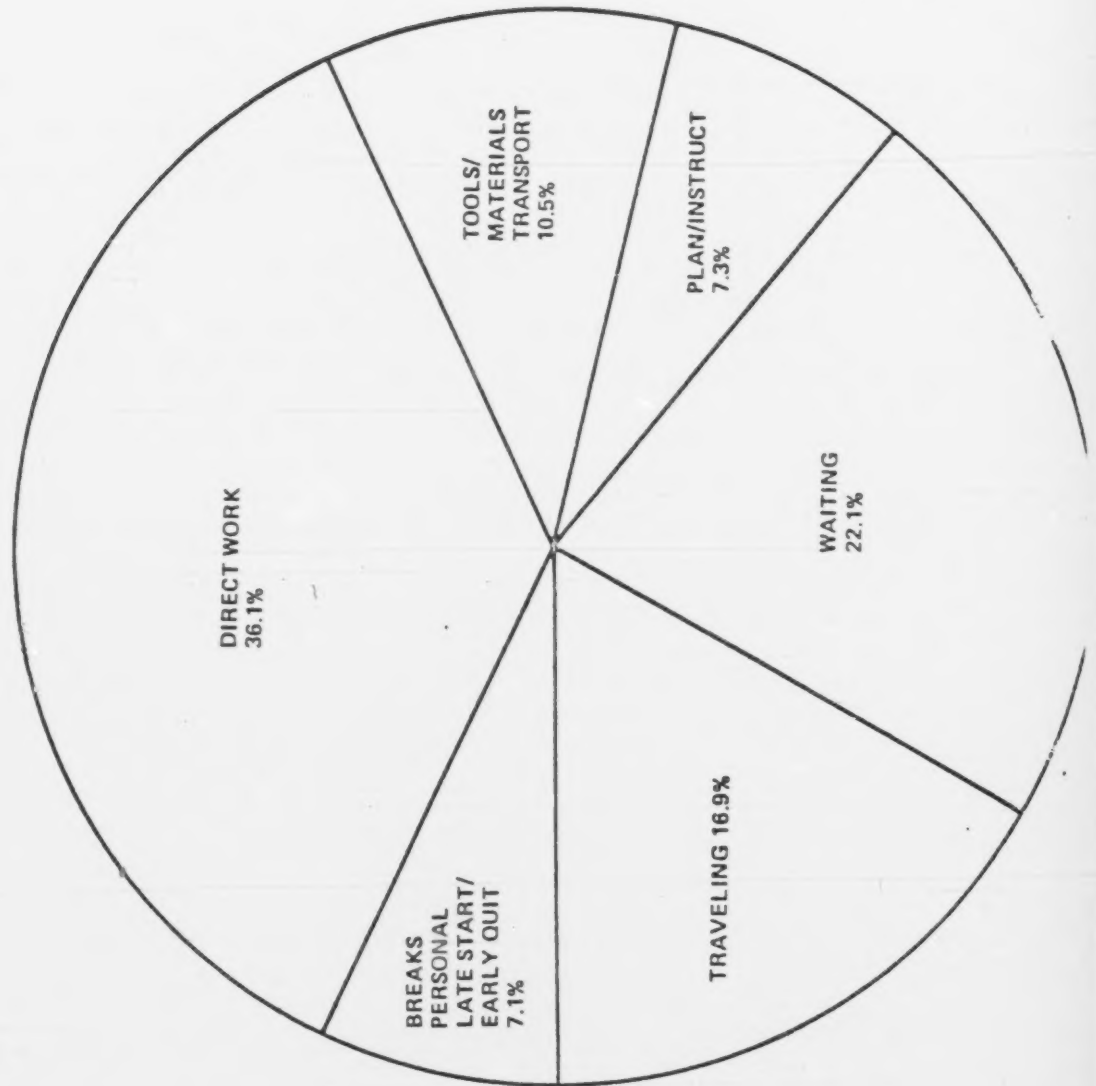
the large cost escalation. Most of the escalation seem to be attributable to the field work, huge increases in manhours, and the fact that construction time is prolonged. The cost of financing construction, ie. the allowance for funds used during construction, also adds significantly to the total cost escalation. If the total plant were designed and sold on the same basis as the NSSS and turbine, then it appears that total cost escalation would be easier to control.

Nuclear power plants require the use of the highest level of modern technology and therefore they cannot be constructed by using techniques that would apply to construction of commercial buildings. One would therefore question the use of "fast track" field construction where engineering design barely leads construction activities. To keep costs down, it would appear that the major portions of the engineering of a nuclear plant should be completed before construction begins as is the case for the NSSS and turbine/generator. If engineering does not substantially lead construction, engineering slowdowns directly translate into schedule delays with corresponding increases in overall cost.

Work sampling at nuclear power plants under construction has clearly demonstrated that field fabrication does not make efficient use of manpower. The percentage of the direct work time of a craftsman at a nuclear construction project has been observed to average approximately 35% as shown in Figure 2, from Reference 8. A recent work sampling by Theodore Barry & Associates⁽⁹⁾ confirms this number and further shows that direct time spent on field fabrication of piping is often significantly less than this average. According to Reference 10, piping installation

FIGURE 2

WORK SAMPLING OCTOBER 80 - APRIL 81



represents about 20% of field labor costs for a typical large nuclear plant. It can therefore be seen that there is a real incentive to improve productivity of the craftsmen who build the plant. Budwani⁽⁶⁾ shows that the craft manhours per kilowatt have increased from 6.9 in the early 70's to 19.5 in the early 80's. Crowley and Griffith⁽⁷⁾ give a range of 15 to 33 manhours per kilowatt based on a survey of 16 plants as compared to a reported 6.2 manhours per kilowatt in 1971. Examples are cited by Budwani which show that A/E manhours have undergone similar increases. Modular construction techniques would certainly create a different atmosphere for fabrication of subsystems since a great amount of the work would be accomplished at a facility with more ideal conditions. As stated in Reference 10, piping and instrumentation should be modularized in addition to equipment packages.

Small plants requiring smaller component parts provide incentives for modular subsystem construction techniques. The concept of acquiring packages of components from vendors is not new. Deaerators are purchased with all controls included; feedwater heaters are purchased with control and relief valves included; main condensers were often purchased as a package with other auxiliary equipment such as air ejectors and condensate pumps; turbine generators are supplied with lubricating systems, cooling systems, exciter and even feed pumps. Costs could be kept down if subassemblies of the plant equipment were shipped to the site from the vendors plant in such a state that only a minimum number of mechanical and electrical field connections are necessary. At this time, the major limitation on delivering subassemblies to the site is

the ability of the transportation systems to deliver the product. Thus, the number of subassemblies which can be shop fabricated depends on the maximum size and weight that can be transported to the site. For small plants, therefore, more subassemblies can be shop fabricated. Costs would be reduced since more efficient use could be made of construction manpower due to an increased percentage of direct work time. Furthermore, wage scales are generally lower than those for field work.

Shop fabrication of nuclear power plants has been recognized as desirable and was a key element in the floating nuclear plant developed by Offshore Power Systems. This large plant design offers all, or more of, the advantages of shop fabrication which are available for small plants. However, it is limited to coastal sites. A small plant design could potentially have a higher level of shop fabrication for any site.

In summary, small plants appear to offer the advantage of allowing for shop fabrication of larger numbers of subassemblies. It is also clear that a much greater percentage of the total plant design must be finalized before construction begins. The level of technical expertise that has been applied to the NSSS and turbine/generator must be applied to the whole plant to obtain more stable designs. Small simple plants offer the opportunity to finalize designs early and thus facilitate construction efficiency.

2.1.2 Standardization

Standardization is a likely byproduct of modular subassembly construction since in a shop environment the same drawings and craftsmen would be used. That is, the shop fabrication process provides incen-

tives for standardization in the sense that any differences between units would require constant attention. If the units are identical, then there is less chance of installing a component on the wrong unit. It would very likely be desirable, for example, to design all units for the most limiting seismic conditions at any site, as opposed to using site specific spectra.

It is also clear from the previous discussion on construction techniques, that standardization is needed in the Balance of Plant (BOP) and civil structures since this is where the cost escalation is greatest. Standardization of the NSSS and turbine generator will help control costs to the extent that this allows standardization of the remainder of the plant. Since certain portions of the civil structures, eg. foundations, will always be site specific, standardization will also be aided to the extent that the plant systems and civil structure designs can be separated. This separation would seem to be easier in a smaller plant, since the smaller subassemblies or modules may be easier to design independent of civil structure constraints. It may also be easier to incorporate some balance of plant systems into the NSSS or turbine generator modules to reduce the number of subassemblies.

To obtain the same total generating capacity a greater number of smaller units would be required. Since the benefits of standardization increase with the number of units, building a greater number of smaller plants would encourage standardization.

2.1.3 Plant Layout and Design

A recent EPRI study surveyed utilities to obtain their input on preferred characteristics of new LWR's. One surprising result was "a fairly widespread perception that the economies of scale may have been pushed somewhat too far and that 1200-1300 MWe plants are well into a size range of diminishing returns". There was some feeling expressed that the larger plants are less conservative and less reliable; maintenance takes longer and replacement power cost penalties are too high.

In addition to the above general comments, a number of specific design improvement suggestions were made by the utilities. Many of these would appear to be easier to implement in a new generation of small plants. Included among the suggested improvements are:

1. single function and separated safety systems,
2. enlarged containments for easier access and maintenance and more laydown space,
3. reduced amount of equipment in containment,
4. design for repair and replacement,
5. greater emphasis on operable spares and design redundancy,
6. emphasis on fail safe and passive designs,
7. improve recirculation pump seals,
8. increased fuel cycle length,
9. design to slowdown transient response. .

Several of these areas are worth discussing in more detail.

Containment size and work space can be increased. However, for the same size plant this will raise costs. There is a tradeoff in the sense that

a larger containment can, for the same NSSS, be designed to a lower peak pressure. For a dry containment, the peak pressure reached following a LOCA will decrease as containment size is increased. The design pressure must be above the LOCA peak pressure. Perhaps, the message here is that utilities who operate plants would prefer larger containments. There would be some cost increase, but this would not be proportional to size since the design pressure could be decreased.

The size of containment required depends on the amount of stored energy in the primary coolant system rather than on the power rating of the plant. The desire to slow down the transient response can be achieved by increasing the primary system inventory. However, this will increase the demands on the containment. Lowering the power density will also slow transient response somewhat, but this would require either derating or a larger volume core. The latter would also increase primary coolant inventory to some extent.

Some early smaller plants have large conservatively designed containments. Economic pressure to reduce the amount of materials and, hence, the size of containment, have led to cluttered containments of sophisticated design, including in some cases ice condensers and pressure suppression chambers. Since materials costs are not the primary cause of price escalation, perhaps reconsideration should be given to either enlarging containment size or reducing plant size.

Longer fuel cycle length will require that more fuel be placed in the core or that higher burnup be achieved. One possibility is to use a large core in a small plant. That is, a large vessel and core with

small plant components. The core would be run for a longer time at a lower power density. This would slow down transient response in addition to extending the fuel cycle. An added benefit would be the reduced fast neutron fluence on the pressure vessel. The reduced size of the rotating equipment in the smaller plant would also facilitate extending inspection intervals, another requirement for lengthening the fuel cycle.

Pump seal problems were also mentioned in the utility survey. The large size of current generation pumps may be contributing to these problems. Smaller pumps would likely help the situation. One way to reduce the size of pumps on a PWR would be to use jet pumps, instead of the present large circulating pumps. In addition to reducing the size of rotating equipment, this would enhance natural circulation capability.

Some of the small plant benefits and features desired by utilities could be obtained in a large plant by utilizing multiple fractional capacity components (eg. two half capacity pumps rather than one full capacity pump). This would provide for more operational flexibility and facilitate repair and replacement. One could even consider two half capacity turbine generator sets, given that these components are now a very small part of overall plant cost.

In summary, utilities who operate the plants are providing feedback which indicates a desire for more design simplicity and more attention to operational and maintainance considerations during design. The desired simplicity would seem to be easier to achieve in small plants.

2.1.4 Operational Considerations

There are those who hold that an operator can have a better feel for a small plant. That is, he knows the plant better and can better anticipate system responses. Greater design margin in small plants could also be a factor. However, it is very difficult to establish quantitatively that small plants are easier to operate than large plants, since many large plants are operated very successfully.

In Section 2.2.2, precursors to severe core damage accidents are considered for comparable small and large plants. The number of samples is very small, and hence one can only make observations based on the data rather than draw conclusions. Based on this data, it appears that although the frequency of precursors is about the same for large and small plants, the severity of precursors appears to be much greater for large plants. This suggests that the small plants are better controlled in upset situations.

It should be noted that the response time of a plant depends on its design and not on its size. The design parameters which influence response time include power density, coolant inventory, circulating pump inertia, maximum loop piping size, etc. Therefore, any differences in the ability to control a small plant are not due to inherently different response characteristics for small plants. The feedback from utilities seems to indicate a desire to slow down system responses. This requires lower power density and/or greater system coolant inventory both of which would be easier to achieve in a small plant.

Load following capability is another feature which utilities identified

as a preferred characteristic for future LWRs. While there is no inherent advantage in this area for small plants, the need for load following would be reduced somewhat by the greater flexibility offered by several smaller units when compared to one large unit. That is, several small plants can operate very efficiently at a number of different total capacity levels.

2.1.5 Maintainability

Reducing reactor size to a greater extent than containment size will improve the environment for maintainance and repair. If smaller components are used (ie. fractional capacity components) space required for dismantling the equipment (pull space) will also decrease. Implementing a modular construction concept could in fact require more space for maintainance and repair if the entire module is replaced.

The use of mockups for maintainance training would be facilitated by smaller plants since the mockup itself would be smaller and it could be used for many plants. Spare parts stocking would be made easier due to the smaller size of parts plus the need to store less parts per unit at multiple unit sites. Since smaller parts would likely require less manhours to change out, the amount of time spent in critical areas should be reduced. Also, scheduled outage time could be reduced.

While changeout of major components, such as steam generators, should not be required in future plant designs, some provision should be made to accomodate the unexpected. Even if the component lasts the life of the plant, life extension may be an option to be considered,

and change out may be desirable at that point. The ability to change-out major components would be much easier to provide in a small plant.

2.1.6 Reliability and Availability

There are many factors which affect the reliability of a nuclear power plant including plant age, regulatory climate, operational philosophy, etc. Availability, ie. the percentage of time a plant is available to generate electricity, is one measure of plant reliability. There are a number of potential complicating factors, which make it difficult to determine the effect which plant size has on availability. These complicating factors include:

- the different time frames, and subsequently the different regulatory and public acceptance climate, during which small plants were built.
- different levels of technical expertise among the utilities who operate nuclear plants.
- differences in design due to the use of different NSSS vendors, A/Es and constructors.
- design improvements made in plants constructed later in time.
- possible different operational philosophies for small plants.

While it is recognized that these factors influence plant availability, it is nevertheless useful to consider the historical data on plant availability as a function of plant size. To reduce the influence of plant

age and time frame as much as possible, availability comparisons were made for small (< 700 MWe) and large (≥ 700 MWe) plants which came on line during the same time period. Figure 1 shows that during the 1968 to 1975 time period, both small and large plants began commercial operation. Table 1 lists the sixteen small plants and thirty-one large plants which began commercial operation during this period. Figure 3 shows average availability as a function of years after start of commercial operation for both the large and small plants. Figure 4 is a similar plot as a function of calendar year. The availability data were obtained from Reference 11, while the power rating and date of commercial operation are from Reference 12. It is apparent from these plots that the small nuclear plants have an average availability which is perhaps 5% greater than that of the large plants. This is a consistent trend which shows up when the data are plotted either on the basis of plant age (ie. years after start of commercial operation) or calendar year.

As stated earlier, there are a number of complicating factors which could partly account for the greater availability of the small plants. For example, it could be argued that the more technically competent utilities got into the nuclear power generation business earlier when the small plants were available. However, use of data for plants which came on line during the same period should minimize any bias due to this factor.

It is interesting to note that the availability data for fossil plants also support the argument that smaller units are more reliable.

TABLE 1. LIST OF NUCLEAR PLANTS WHICH BEGAN COMMERCIAL OPERATION

1968 THROUGH 1975

LARGE PLANTS			SMALL PLANTS		
PLANT	MWe (net)	Yr. of Comm. Oper.	PLANT	MWe (net)	Yr. of Comm. Oper.
Dresden 2	809	71	San Onofre	440	68
H. B. Robinson	707	71	Connecticut Yankee	580	68
Dresden 3	788	71	Oyster Creek	650	69
Palisades	822	72	Nine Mile 1	625	69
Quad Cities 1	789	73	Ginna	470	70
Quad Cities 2	789	73	Millstone 1	660	70
Surry 1	822	72	Point Beach 1	497	70
Turkey Point 3	725	73	Monticello	548	71
Main Yankee	825	72	Vermont Yankee	514	72
Surry 2	822	73	Point Beach 2	497	72
Oconee 1	886	73	Pilgrim 1	670	72
Indian Point 2	864	74	Fort Calhoun	478	73
Turkey Point 4	725	73	Prairie Island 1	507	73
Zion 1	1040	73	Kewaunee	540	74
Brown's Ferry 1	1067	74	Arnold	538	75
Peach Bottom 2	1051	74	Prairie Island 2	507	74
Oconee 2	886	74			
Zion 2	1040	74			
Cooper	778	74			
TMI 1	820	74			
Brown's Ferry 2	1067	75			
Peach Bottom 3	1035	74			
ANO-1	850	74			
Oconee 3	850	74			
Rancho Seco	928	75			
Hatch 1	797	75			
Calvert Cliffs 1	865	74			
Fitzpatrick	821	75			
Cook 1	1054	75			
Brunswick 2	821	75			
Millstone 2	870	75			

FIGURE 3 AVAILABILITY VS PLANT AGE

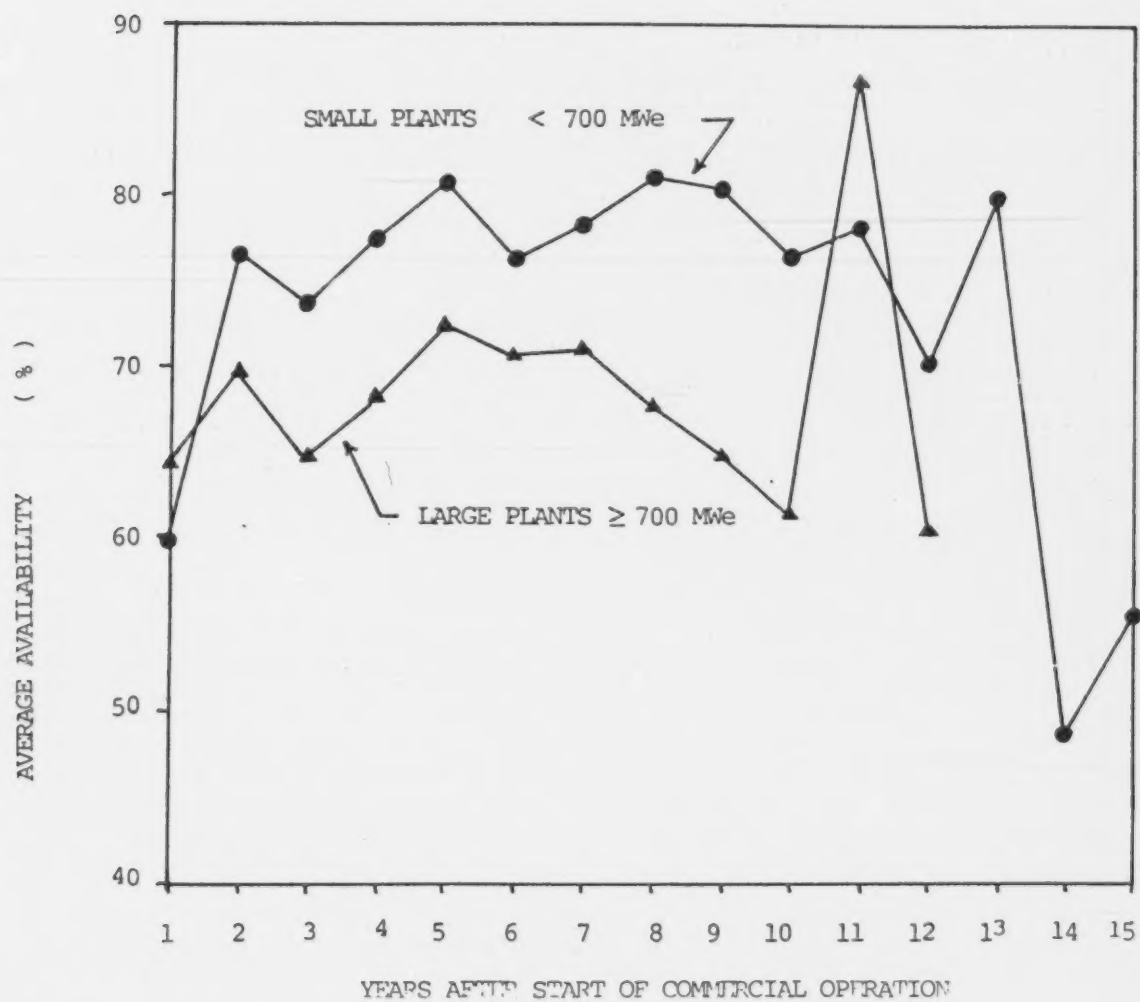


FIGURE 4 AVAILABILITY VS TIME

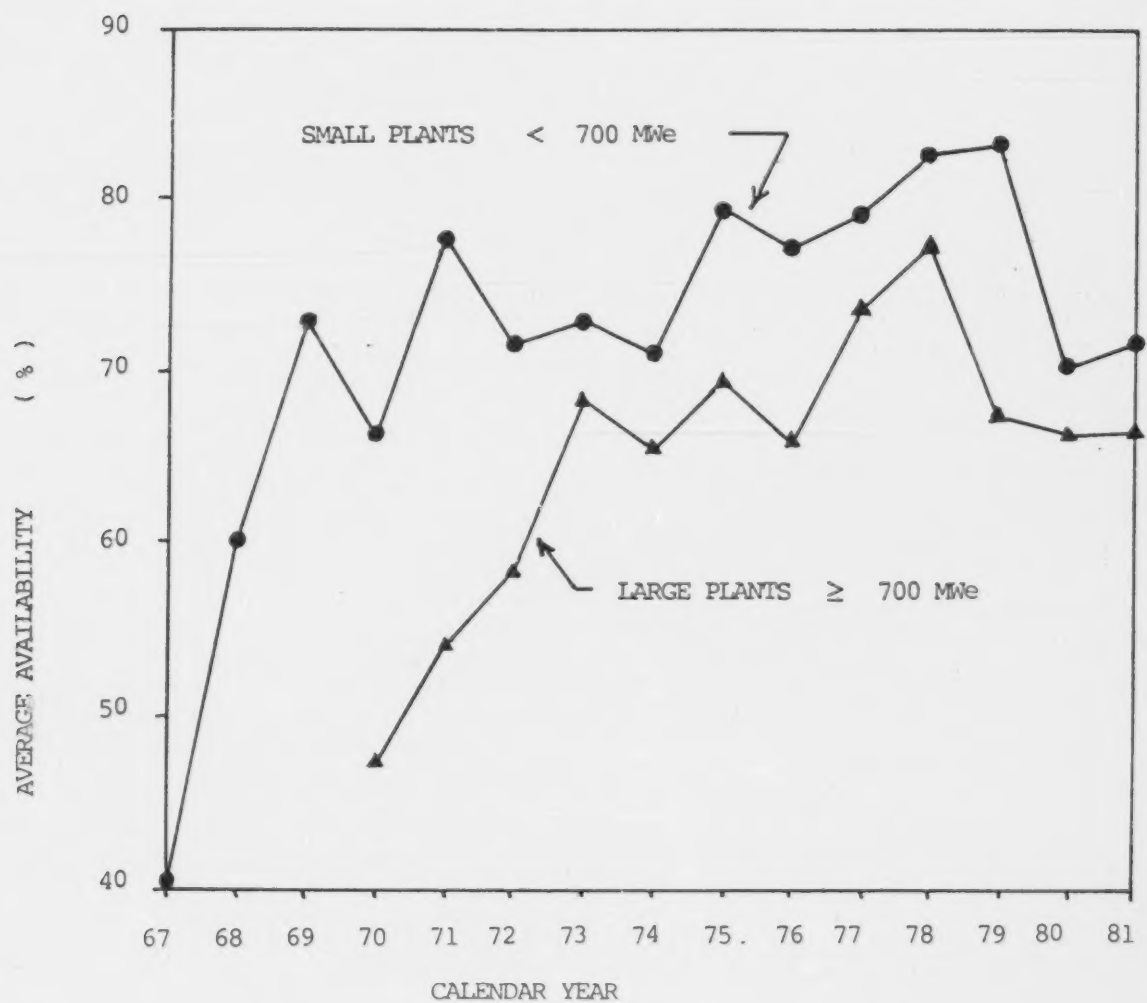


Table 2 shows National Electric Reliability Council (NERC) availability data⁽¹³⁾ for the period 1971-1980 for various size fossil steam cycle plants. The trend of increased reliability for smaller units is evident. Figure 5 shows a plot of largest U.S. turbine generator ordered for fossil and nuclear units through 1976. While the size of the largest nuclear units ordered during this period increased, the trend to increased size seems to level off or decrease for fossil units.

The reasons for this trend of higher reliability for smaller units could be due to a number of causes. It could be that the forced outage rate is lower for smaller units, or it could be that the forced outages are shorter in duration. That is, it may be easier to get a smaller unit back on line. It is also possible that the scheduled outages are shorter due to easier maintainance. A much more detailed study of the outage data would be required to gain further insight in this area.

The historical data show rather convincingly that small plants have a higher availability on average, than large plants. Underlying causes for the higher availability of small plants would be very difficult to establish, especially since there are some large plants with availability comparable to that of small plants. In addition to the apparent economic implications of this higher availability for small plants, one can ask what, if any, implications this higher availability has in other areas. For example, if this higher availability is due to easier operation and maintainance of small plants, then what does this imply about accident risk? If small plants are easier to operate and maintain, then does ownership represent a smaller financial risk? On the other hand, it

TABLE 2 STEAM CYCLE FOSSIL UNIT AVAILABILITY (1971 - 1980)

<u>UNIT SIZE (MWe)</u>	<u>AVAILABILITY</u>		
	<u>UNIT</u>	<u>TURBINE</u>	<u>BOILER</u>
200 - 299	83.6	93.4	87.2
300 - 399	79.5	91.1	84.6
400 - 599	77.4	90.3	84.8
600 - 799	76.0	91.3	83.3
800 - above	74.1	90.2	81.5

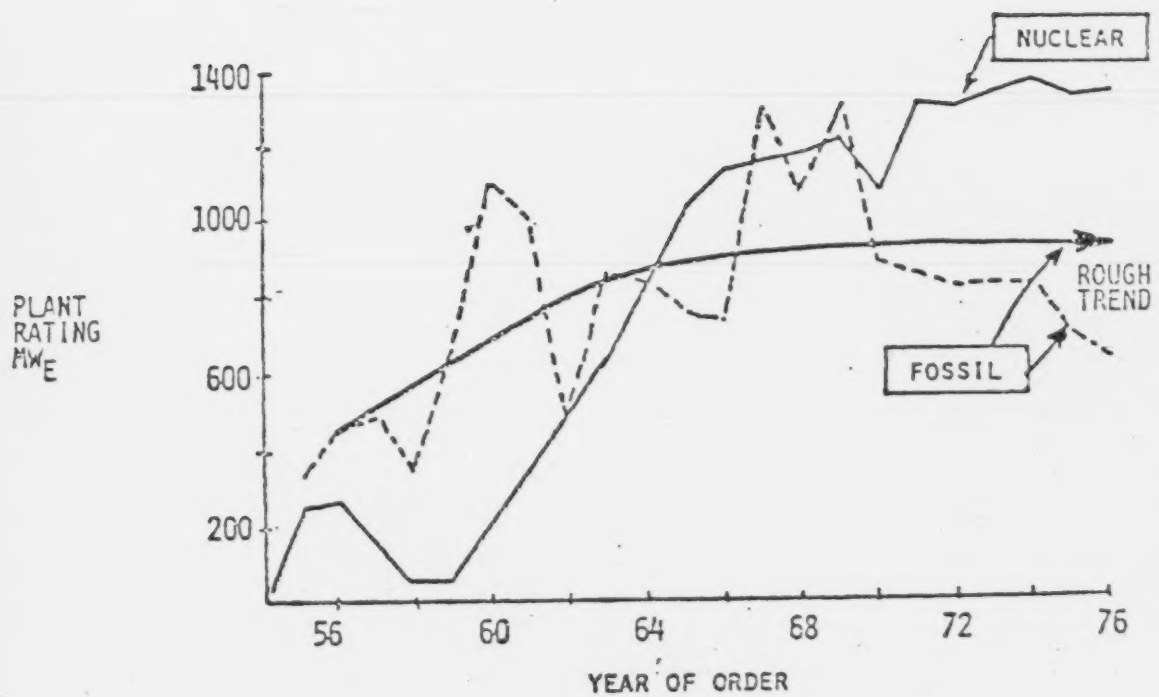


FIG. 5 Largest U.S. Turbine/Generator Ordered

could be that plant operators are overly cautious with large costly plants and tend not to push them as hard, in which case they would represent less of a financial risk.

2.2 Safety Factors

When discussing safety, it is convenient to separately consider Design Basis Accidents (DBAs) and Severe Core Damage Accidents. A third safety factor to be considered is plant siting.

2.2.1 Design Basis Accidents

Safety systems are installed on nuclear plants to protect the plants during anticipated operational transients and a variety of postulated accidents. The plant performance, ie. maximum power level, maximum peaking factor, etc., will be limited by the ability of the safety systems to keep plant response within prescribed regulatory limits. Plant performance analyses for Design Basis Accidents (DBAs) are contained in Final Safety Analysis Reports (FSARs). Some of these analyses are repeated for each reload.

Based on the calculations performed for licensing purposes, it is fair to say that there is no inherent dependence of DBA response on reactor size. Rather, accident responses depend upon plant design features such as core power density, number and size of coolant loops, safety system performance, etc. Early DBA sensitivity studies performed by Westinghouse⁽¹⁴⁾ indicated only small differences in plant performance limits for their various size PWR plants, even though there were some design

differences (number of coolant loops). Subsequent analyses have demonstrated that current safety systems will protect either small or large plants during DBA events.

Certain improved safety features have been proposed for new LWRs and there is a renewed interest in inherently safe designs. We might ask whether any of these features, eg. passive decay heat removal systems, are easier to install on small plants. For an LWR, a passive decay heat removal system would normally require that a large source of water be made available to flow into the primary system whenever there was the potential for voiding the core. Although a large supply of cooling water would be needed for a large plant, there is no inherent advantage in this regard for smaller plants. Another proposed safety improvement is to connect the pressurizer surge line to the upper head instead of to the loop piping as is the present practice. This would prevent voiding of the upper plenum while there is a level in the pressurizer. Large volume pressurizers have also been considered to slow down transient response. Again, the small plant does not have any overwhelming advantage as regards these improvements. In fact, a number of such safety improvements are planned for the U.K. Sizewell B unit and the Westinghouse APWR which are both large plants.

The one design safety change which could possibly be easier to implement in smaller plants is a decrease in core power density. This would slow down transient response and could also improve the prospects for using passive safety systems. For example, a lower power density core could be used in a large plant vessel with smaller loop components.

The above items are representative of evolutionary design changes. More revolutionary inherently safe designs are being proposed almost exclusively in conjunction with new small reactor designs. These are discussed in more detail in sections 5 and 6. Presumably, some of these concepts could also be applied to large reactors. However, proof testing would be more straightforward on small plants.

2.2.2 Severe Core Damage Accidents

Present reactor designs require that some portion of the engineered safety systems function in order to prevent severe core damage for various accident initiators. While the probability of multiple safety system failures is very remote, the consequences of such failures can be very severe also. The risk to the plant and to the public safety is a function of both the probability of occurrence and the consequences. It is of interest to consider whether there is any basis for believing that either severe accident probabilities or severe accident consequences are very different for small or large plants.

2.2.2.1 Accident Probabilities

There are a number of events which have occurred at operating nuclear power plants which could have led to serious accidents had safety systems failed to operate, or had the operators failed to take corrective action. There have also been instances where safety systems have been found to be inoperable during testing or maintenance. Licensees of nuclear plants are required to file Licensee Event Reports (LERs) with the NRC for such

occurrences. Initiating events which require operator action or the performance of safety systems can be regarded as severe accident precursors in the sense that they could have developed into more serious incidents.

Following the TMI accident, a strong interest developed in reviewing and evaluating LERs, since several incidents had occurred prior to TMI which were regarded as precursors to TMI. Oak Ridge National Laboratory has conducted a review which attempted to identify significant accident precursors which occurred at domestic plants between 1969 and 1979. LERs were screened and evaluated to identify severe accident precursors. Reference 15 reports on the results of this work. A total of 169 events were identified as severe accident precursors. Each was assigned a severity classification based on the probability of the event leading to severe core damage. Of the 169 events, 52 were identified as contributing significantly to the overall risk of severe core damage. Statistical analysis of the data is very difficult due to this small sample size. Some small sample statistical analyses were conducted without distinguishing severity among the 52 precursors. On this basis, no statistically significant dependence on power level, vendor, A/E or plant type were found. These results indicate that the frequency of occurrence of precursors is independent of plant power rating (ie. size), and the other factors listed above.

The Nuclear Safety Analysis Center (NSAC) has screened and evaluated LERs for 1979,⁽¹⁶⁾ 1980^(17,18) and the first half of 1981.⁽¹⁹⁾ Their findings also do not suggest any dependence of the frequency of events on plant size. For example, two of ten "most significant" events

in 1979 and 10 of 62 significant events in 1980 occurred at small plants.

It is interesting to note, however, that in the ORNL study, 82% of the risk comes from just three initiators: the TMI accident, Brown's Ferry cable tray fire and Rancho Seco non-nuclear instrumentation failure. Note that all of these are large plants. Several initiators have occurred since 1979 at small plants, ie. Ginna and Prairie Island steam generator tube ruptures, which could change this situation. However, the present data suggest that initiating events at small plants may be easier to manage than similar events at large plants. Proper handling of initiators at Prairie Island and Ginna tend to confirm this conjecture. ORNL is continuing its study considering initiating events which occurred during 1980 and 1981. It is noted that the Probabilistic Risk Assessments (PRAs) conducted to date (WASH-1400, RSSMAP and IREP programs) have considered primarily large plants. However, the limited work (Millstone 1 and Big Rock Point PRAs) do not indicate that there is a size dependence in the probability of core melt.

2.2.2.2 Accident Consequences

It is clear that the worst accident consequences are proportional to the amount of fission product inventory stored in the plant. For similar fuel cycles, the fission product inventory is roughly proportional to power rating. Therefore, the worst accident consequences for three one-third capacity units at one site would be approximately one-third those for a full capacity unit.

Since there is only a very small data base on severe core damage

accidents, the physical processes involved in core melt and fission product release are not well understood. It has been conjectured⁽²⁰⁾ that core melt accident consequences are now significantly overpredicted. Probabilistic risk assessments conducted to date are almost exclusively for large plants. The methodology used in PRA's assumes that core melt occurs when certain combinations of system failures or operator actions occur. Hence the probabilities of core melt are dependent on the system reliability rather than on phenomenological analysis. It is apparent, however, that the central part of a core will begin melting before the periphery, due to the presence of structural heat sinks near the periphery and the higher power density near the middle of the core. This is in fact what occurred at TMI. For a smaller core, a higher fraction of the core is located near the peripheral heat sink. Hence, for the same accident sequence, one would expect a smaller fraction of the core to melt. A rough measure of the ability to radiate heat to the peripheral heat sink is the ratio of the outside surface area of the core to the total power generated. This quantity is higher for a small core. Current consequence analysis techniques model this in only a very approximate sense.

The fission product release could also be reduced for a small plant in a large containment, since a larger amount of surface area would be available for plate out of fission products. One would also expect containment failure probability to be reduced for this situation.

2.2.3 Siting and Licensing

Siting of nuclear plants has become much more conservative in that future plants will certainly not be built as near to population centers as some present plants. The question of how near depends in part upon predicted maximum off-site accident doses. Since these would be lower for a small plant, or for multiple small plants on a site, it would appear that small reactors have an advantage over large reactors in terms of the number of suitable sites. Concentration of nuclear generating capacity in energy parks has been suggested since this has certain advantages in terms of operational support, security, etc. Such a trend would not necessarily work against small plants, since they provide flexibility in achieving the maximum power generating capability at a site. The Canadian experience (up to eight small reactors on one site) is worth noting in this regard.

Ease of licensing should depend on plant design rather than size. However, in the past, regulators have appeared to focus their efforts more strongly on large plants since these constitute the greater total capacity and therefore the greater overall risk. Licensing would likely be made more tractable for plant designs which have greater inherent safety characteristics. To the extent that small plants could be designed to incorporate more inherent safety features, they should be easier to license.

2.3 Cost Factors

2.3.1 Capital Cost

A sensitivity study of capital costs as a function of unit size is reported in Reference 21. The analysis assumed that all other parameters except unit size are held constant including design and construction period and year of first commercial operation. Although the dollar figures of this 1979 study may no longer be accurate due to escalation, the trend information is useful. Figure 6 shows the unit capital cost vs. unit size for a single unit LWR and a comparable coal plant. These results indicate that the economies of scale favor large plants. However, the assumption of equal design and construction time may not be appropriate. The construction time for the small plant could be significantly less, especially if modular construction techniques are used.

Figure 7 shows estimated unit capital cost for small and large plants built during the 1968-1975 era when both size plants were being built. The plant to plant cost variations tend to mask the economies of scale, but a slight average capital cost advantage for large plants is indicated. Since the data do not show a clean economy of scale dependence, other factors (project management expertise?) may be more of an influence on cost than economy of scale.

2.3.2 Operations and Maintenance

Reference 11 gives the average Operating and Maintenance (O&M) cost for public and investor owned nuclear plants in 1981 as 6.30 mills/

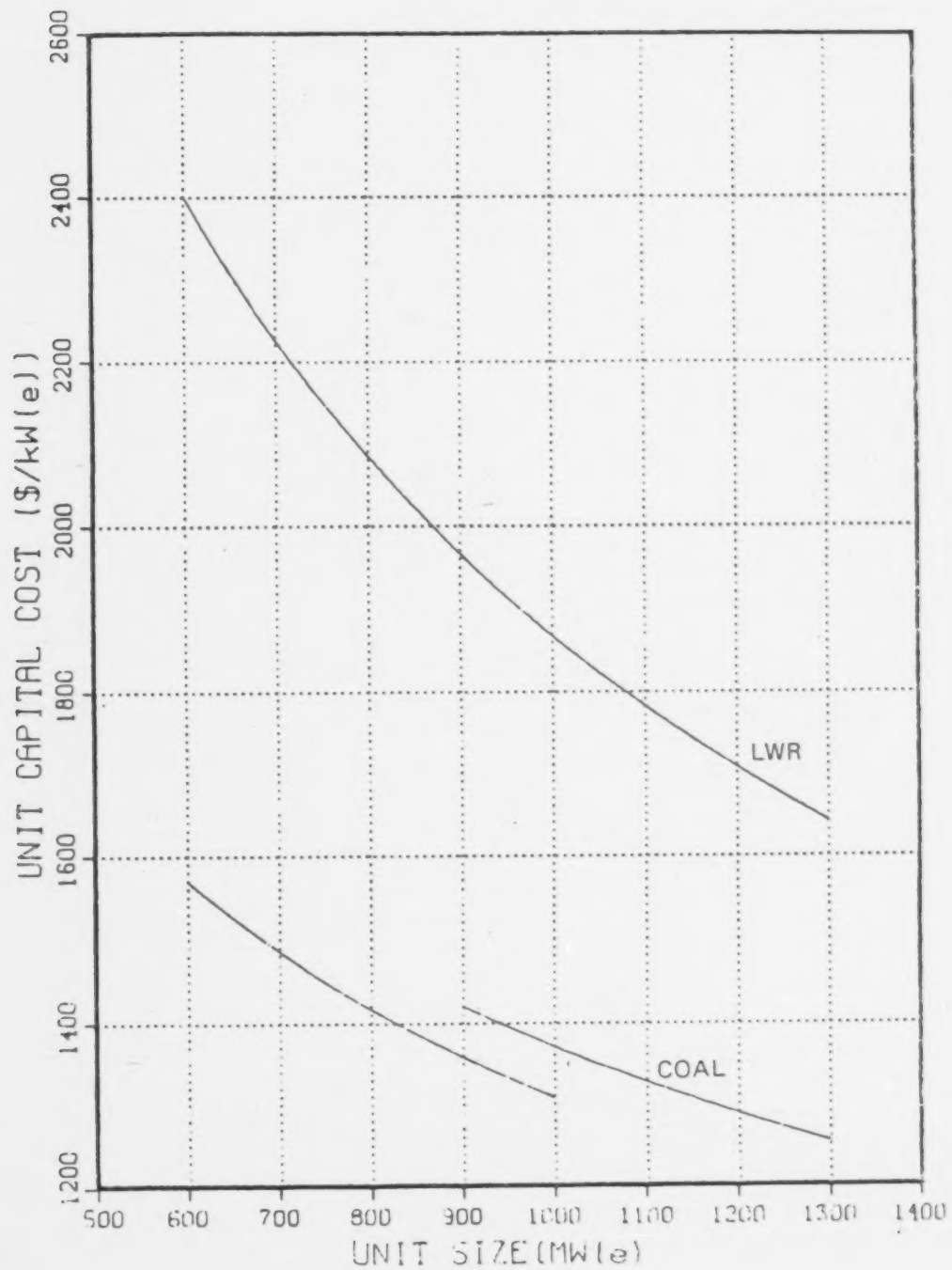
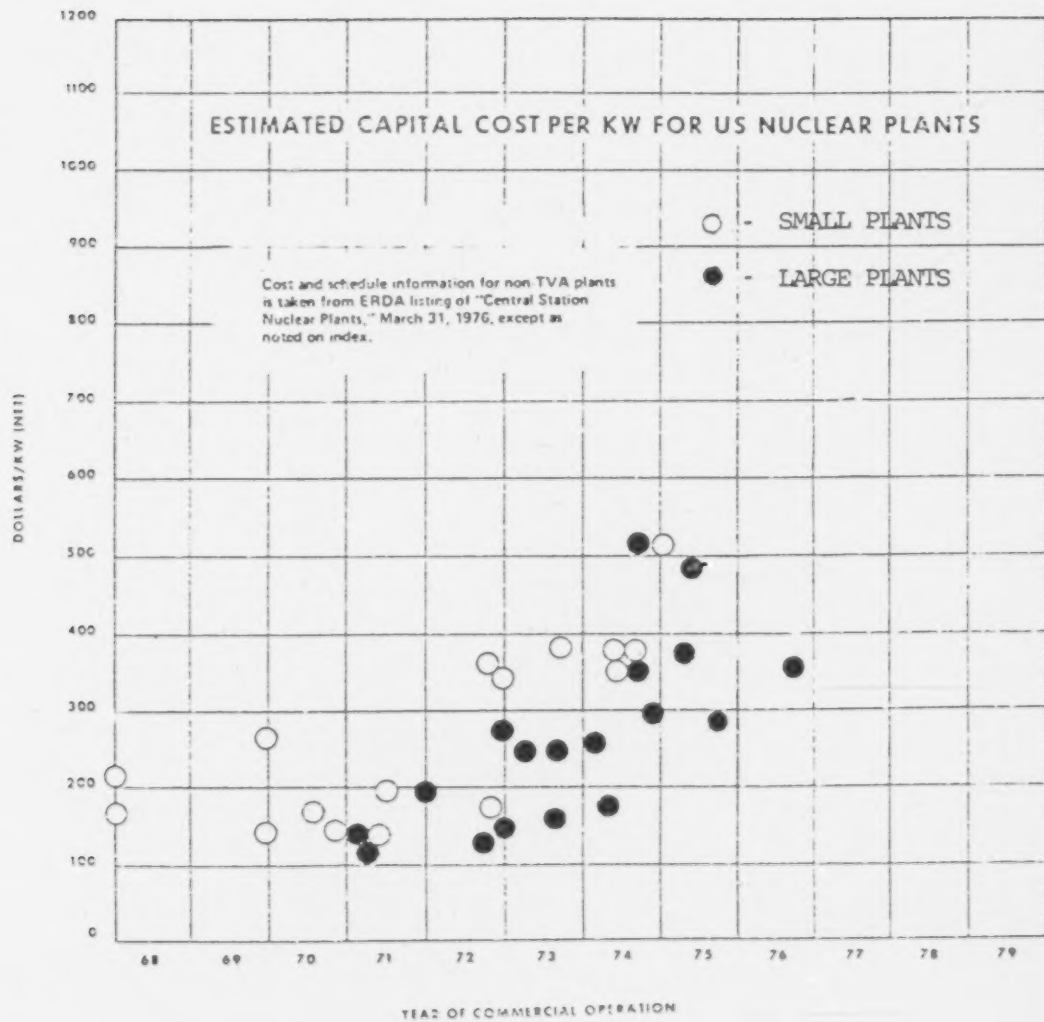


Fig. 6 Estimated power plant capital investment costs as a function of unit size for 1990 operation.

FIGURE 7



Source: TVA document

kWh. Perusal of the individual plant cost data in Reference 12 shows a large variation which totally masks any advantage in this area for large plants. One might expect an O&M cost advantage for large plants since such costs as staff and security would be expected to be more or less independent of size. This advantage, however, is not obvious from the referenced data.

2.3.3 Fuel Cost

Reference 11 gives the average fuel cost for public and investor owned nuclear plants in 1981 as 5.44 mills/kWh. Data given in Reference 12 show considerable scatter which again masks any clear advantage that large plants may have in this cost area.

3.0 ECONOMIC CONSIDERATIONS

3.1 Components of Cost

As discussed in the preceeding section, O&M and fuel unit cost may be slightly lower for larger plants. Significant trends of these costs as a function of plant size are masked by plant to plant variations. The major cost component for a nuclear plant is capital cost. Careful consideration of this component is therefore justified in comparing small and large plants.

3.2 Large Plant Cost

Large plant construction costs have increased so rapidly that it is extremely difficult, if not impossible, to accurately estimate the cost of a plant on which design and construction are started now. As an example, VEPCO's North Anna 3 plant was originally estimated to cost \$301 million. The latest cost estimate, at the time the project was cancelled was \$5.1 billion. The cost estimate had doubled in the last two years. A more detailed analysis of nuclear plant cost overruns is given in Reference 22.

Some utilities are in the position of spending hundreds of millions per year on construction of nuclear plants, only to learn that the cost estimate to completion at the end of the year is greater than at the beginning of the year. With this situation, it is not surprising that plants with significant construction progress are being cancelled.

Reference 23 estimated the cost of construction of an 1100 MWe nuclear plant for which construction is started now with a 1993-1994

date of operation. Four cost elements were projected and estimated individually. These are:

1. Direct costs - materials and equipment at today's prices
2. Indirect costs - engineering, construction management, quality assurance, permits, etc.
3. Allowance for Funds Used During Construction (AFUDC) - interest on money used during construction
4. Escalation - allowance for inflation of prices

Unit costs were found to be very sensitive to the assumed inflation rate and ranged from a low of \$2519/kw to \$8227/kw. This corresponds to a total capital cost between \$2.77 billion and \$9.05 billion.

These values bracket an estimate of \$4.3 billion for a 1000 MWe plant given in Reference 7.

3.3 Small Plant Cost

If a small plant were to be built starting today using the same methods and construction techniques used for large plants, the unit construction costs (\$/kw) would very likely equal or exceed those of the large plant. This conjecture is based on the data presented in Figure 7.

However, innovative construction techniques could be used to make small plants economically attractive. As an indication of what might be achieved, it is interesting to consider the projected cost of some novel designs. Reference 24 considers the economics of a two-unit Consolidated Nuclear Steam System (CNSS) compared to a two-unit HTGR

and a two-unit coal fired plant. While the analysis is for a cogeneration facility at a large oil refinery, the construction costs should be similar for an electric generating facility.

The two-unit plant would have a capacity of about 800 MWe. Significant economic benefits are assumed due to a shortened construction schedule, and the liberal use of shop fabrication techniques. The estimated total base capital cost for the two-unit plant is \$886.5 million in January 1980 dollars. An estimate was also made for the two-unit HTGR of \$1.246 billion.

4.0 INSTITUTIONAL CONSIDERATIONS

In addition to technical and economic factors, there are also institutional considerations which could impact on reactor size. These considerations basically deal with the way utility managers, regulators and the general public view the plant size issue.

4.1 Utility Perspectives

The utility industry is a diverse network of public and investor owned organizations, so it is difficult to generalize utility perspectives on any subject. However, the current mood might be characterized as an openness to consideration of the potential benefits of small reactors.

A recent survey of utility personnel by EPRI found that an interest in smaller reactors was brought forth by the utilities themselves. Most of the utility personnel interviewed were clearly associated with plant operations as opposed to generation planning. Their interest in small reactors is likely based on technical rather than economic considerations. However, utilities also perceive that small reactors would be more compatible with present load growth projections and could also minimize the sometimes burdensome practice of fractional plant ownership.

Another consideration is the time period required to bring a plant on line. The current twelve year time frame for construction of a new nuclear generation facility is beyond the point where most utilities can plan with any degree of certainty. If small plants could be brought on line in a shorter period, then utility interest would likely increase further.

4.2 Regulatory Views

4.2.1 Nuclear Regulatory Commission

The charter of the NRC does not include promotion of nuclear power generation or of specific plant designs. Rather, regulatory criteria are set which plant designs must satisfy in order to be licensed. The only known criteria which deals with size is the 3800 Mwt limit on the rating of a single unit. It would also appear that NRC has focused strongly on the greater number of large units since this group of plants represents a much more substantial contribution to total risk, than the lesser number of small plants.

It was not possible to obtain any statement from NRC personnel on how they would view small plants, ie. favorably or unfavorably. NRC and EPRI are now participating in an industry group which is addressing regulatory stability and design improvements. This group is a follow up to the EPRI effort to determine utility design preferences for new LWRs.

4.2.2 State Public Service Commissions

The charter of state Public Service Commissions (PSCs) is to insure that investor owned utilities provide safe and reliable service. That this service be provided at a fair price, may also be included as part of the charter. In carrying out this charter, a PSC will very likely consider the prudence of utility decisions. For example, a PSC may allow a utility to include the costs of a cancelled unit in the rate base for a period of time, if cancellation is a prudent action on the part of a utility.

Since a unit will not generally be included in the rate base until it is operational, some cancellations may have been influenced by rate base considerations in addition to load growth forecasts.

Once a utility has cancelled a large unit after incurring significant costs, it will be difficult for a PSC to allow for cost recovery on any new large plant orders. However, the chances for having to cancel small units are presumably less. For multiple small unit plants, the option to cancel some of the units and retain others also offers prudent flexibility. This argument assumes that the cost of the small unit is substantially less than a large unit.

Siting of plants may be handled by some agency other than state PSCs. However, state agencies would be involved in this process in addition to the NRC. From a technical standpoint siting of a small plant would be expected to be easier than siting of a large plant. From a political standpoint, ease of siting would depend upon public perceptions of risks from small plants.

4.3 Public Perceptions of Risks and Benefits

To some extent public acceptance of nuclear power generation depends on acceptance of nuclear power by the technical community. It is doubtful that small plants would themselves reduce concerns by some members of the technical community regarding the nuclear power generation option. Conversations with a number of intervenors indicated that concerns would remain as long as the consequences of the worst accident are unacceptable regardless of the probability of the event. Inherently safe designs

would be more appropriate for reducing concerns. Transportation risks were also mentioned as a possibly unacceptable hazard. These risks, however, depend only on total generation and not on plant size.

In summary, greater emphasis on inherent safety in plant design rather than plant size seems to be the best approach to reducing technical concerns with the nuclear option. Small plants will reduce these concerns only to the extent that inherent safety features may be more readily included in the design.

5.0 DOMESTIC MARKET FOR SMALL PLANTS

While there is no current domestic market for nuclear plants, utilities will continue to add to their capacity to replace retired units and to meet small increases in demand. The need for new generation is currently being met by fossil (primarily coal) units.

5.1 Utility Needs

Many utilities prefer to have a diverse generating mix of nuclear, fossil and hydro units. The absence of a nuclear option has limited the choice for new base load generation to coal fired units. Nuclear costs has been said to have acted to keep a ceiling on the cost of coal fired units. If this is true, then further increases in the cost of coal units can be anticipated. One would not expect utilities to be pleased with the present situation.

Many utilities are successfully operating nuclear units. The cost of generation for units which came on line in the seventies is very low compared to fossil generation, despite significant capital cost increases during the construction period.

The present situation is such that no prudent utility could order a nuclear unit because of the large uncertainties in cost and delivery. The fact remains, however, that utilities would prefer to have options other than coal-fired generation for base load plants. The current problems with the nuclear option are attributed to regulatory instability and the inflationary economy. Rapid growth of the industry during the boom years

of nuclear power is also a factor to be considered. It is clear that a return of the nuclear option will require some kind of innovative process with the financial risk spread over more than a few utilities. A government subsidized program or a utility consortium approach are possible vehicles for revitalizing the nuclear option.

5.2 Vendor Perspectives

NSSS vendors currently offer 1000 MWe class plants for the domestic and export market. Several vendors state that they have designs for plants from 600 MWe to 1300 MWe. No significant interest in small reactors has been expressed by domestic vendors. There is still a strong belief in the economies of scale and a feeling that small reactors are economically unsound for the buyer and the vendor. Therefore, the only activity in the small reactor field is a few conceptual studies. This is in contrast to a somewhat higher level of activity in this area on the part of other supplier countries.

As discussed in the Introduction, domestic NSSS vendors are concentrating their efforts on improving existing large LWR designs. Emphasis is being placed on improving reliability, extending refueling cycles and on evolutionary safety improvements. Lack of NSSS vendor interest in small reactors appears to be based at least in part on the perception that there is no real market for this product. If such a market were to develop, it is highly probable that several, if not all, NSSS vendors would actively enter the area.

5.3 DOE Activities

The Department of Energy is currently sponsoring a number of studies on small reactors. One study that has been ongoing for several years is the Consolidated Nuclear Steam System (CNSS) work at ORNL. Battelle Northwest Laboratory and Human Affairs Research Centers are assessing the commercial viability of small reactors. Argonne National Laboratory surveyed the market for small reactors in less developed countries and the developments in supplier countries.⁽²⁵⁾ Considerable work is also being done on small HTGR systems.

Perhaps the two most developed new small reactor designs are the CNSS and the HTGR. A brief overview of recent efforts on these systems is given below.

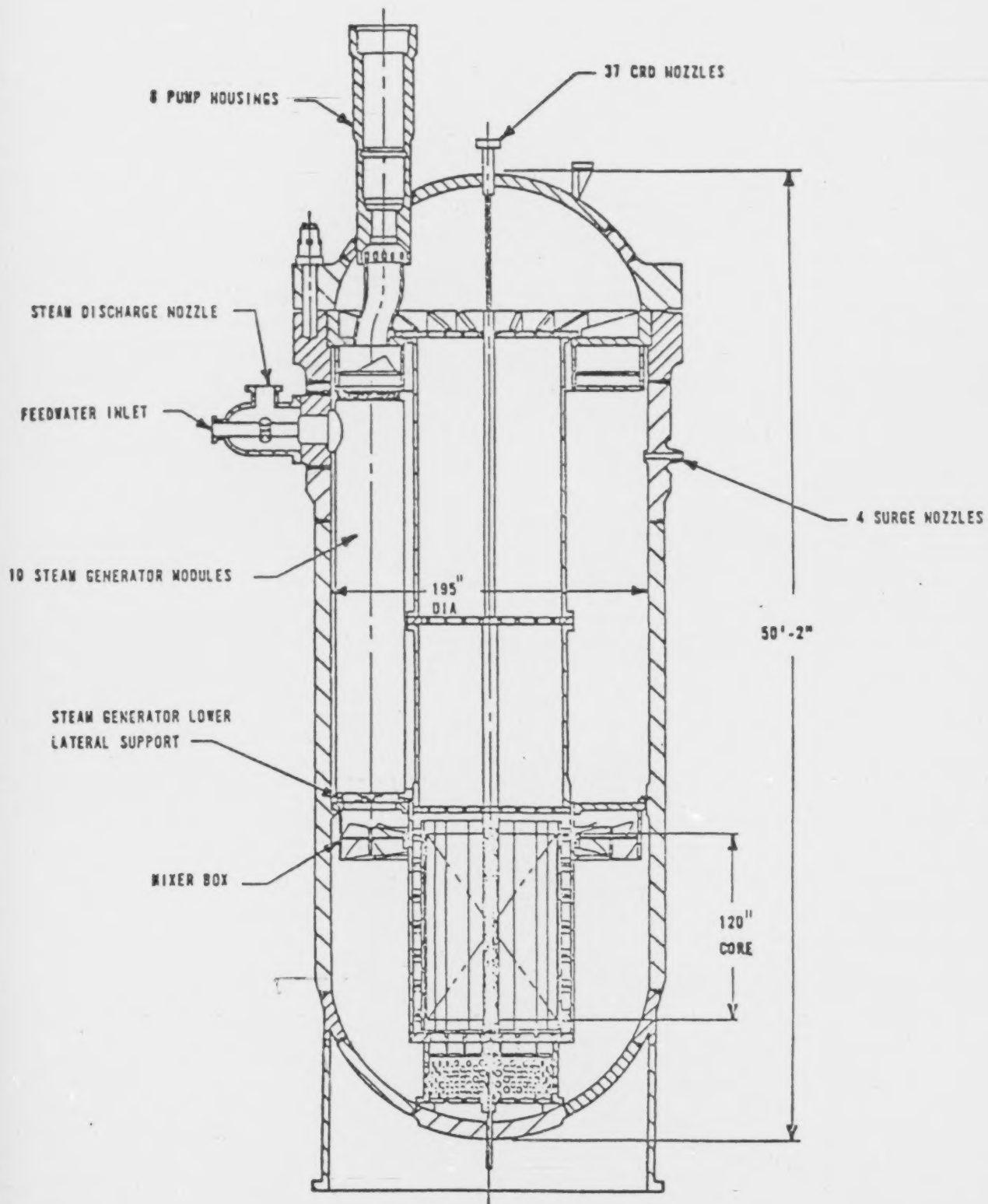
CNSS

The CNSS design⁽²⁴⁾ is based on the Consolidated Nuclear Steam Generator (CSNG) design which was developed for ship propulsion by Babcock and Wilcox under Maritime Administration sponsorship. It is an integral PWR with its nuclear core and steam generators located inside the reactor pressure vessel as shown in Figure 8. The pressurizer is the only NSSS component external to the vessel, being connected by four small diameter surge lines. Eight wet-rotor pumps and ten steam generator modules are located within the vessel which is slightly larger than the vessels used for large PWRs.

The compact design of the CNSS permits shipment of the vessel to the site as a unit, thus facilitating shop fabrication. Preliminary LOCA

FIGURE 8

400 MWe CNSS ELEVATION VIEW



analyses⁽²⁶⁾ performed for a double ended break of one of the four surge lines showed that the reactor core is never uncovered, and that the peak clad temperature is 652°F. Core flood tanks (accumulators) are not required. Thus, the CNSS can be characterized as having good inherent safety features.

HTGR

As stated earlier, the Ft. Saint Vrain HTGR falls into the category of a small reactor. Even smaller designs with additional inherent safety features are currently being developed by General Atomic Co. Although some of the proposed designs fall below 200 MWe in size, the concepts should be applicable to reactors in the 200-700 MWe range. A fairly extensive technology base has been established for HTGRs in the U.S. and other supplier countries. Small HTGR designs are discussed more fully in References 4 and 5.

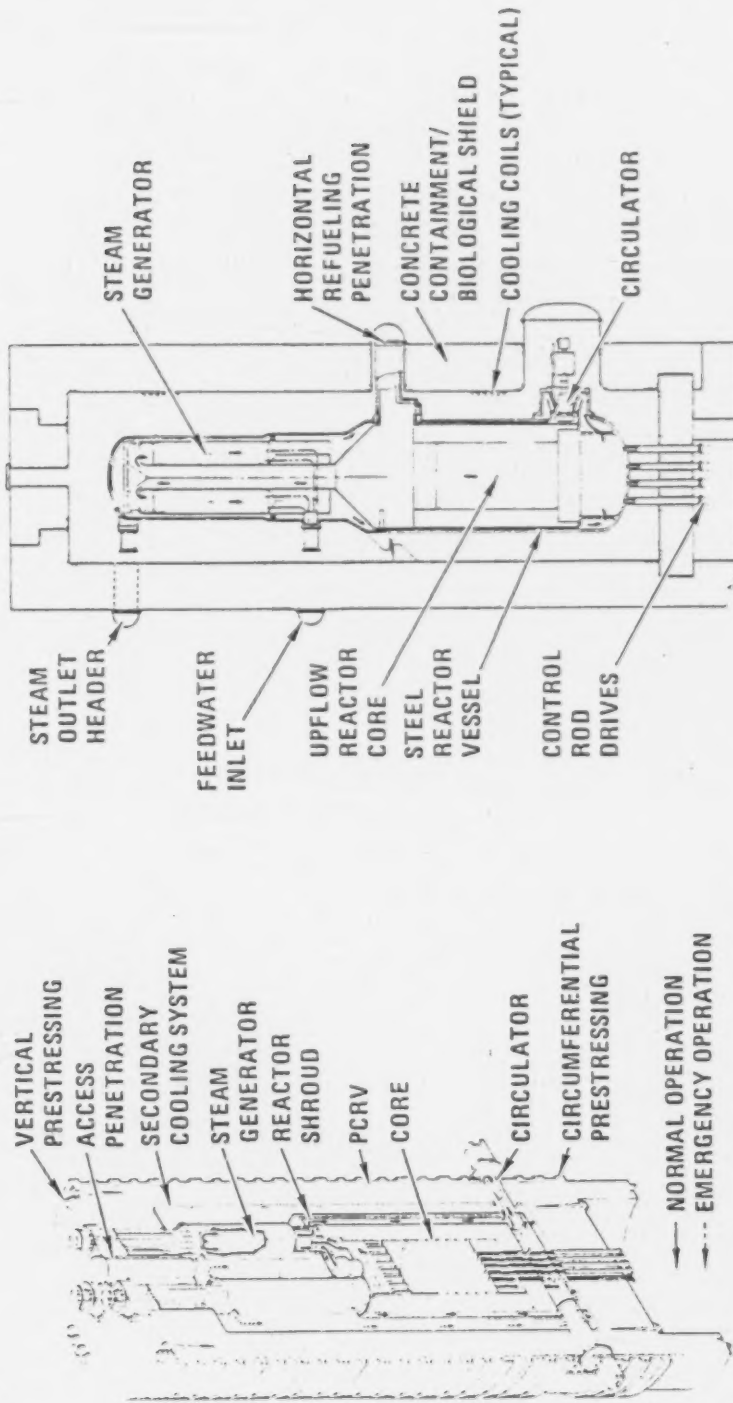
To make the small HTGR economically acceptable, the approach suggested in Reference 5 combines a standardized nuclear heat source, emphasis on a high degree of factory fabrication and system simplicity. Figure 9 shows two small HTGR steam cycle concepts. Inherent safety features are included in the design. The decay heat removal system is such that even under extreme conditions of loss of reactor pressure and loss of forced convection cooling, the fuel temperature will be such that all fission products are retained in the fuel. Even under the most limiting circumstances, ie. multiple safety system failures, the combination of the high core heat capacity, use of a



GENERAL ATOMIC

SMALL HTGR STEAM CYCLE PLANT CONCEPTS

FIGURE 9



PRESTRESSED CONCRETE REACTOR VESSEL (PCR)

- PCR NOT SUBJECTED TO THERMAL CYCLING DURING PASSIVE COOLING MODES
- MAY OFFER ADVANTAGE AS REGARDS CONFINEMENT/CONTAINMENT ISSUE
- PCR CURRENTLY ADAPTED FOR ALL MAJOR GAS-COOLED REACTOR PLANT PROJECTS

ELEVATION VIEW OF MODULAR REACTOR SYSTEM

STEEL REACTOR VESSEL

- OFFERS ADVANTAGE OF FACTORY FABRICATION
- POTENTIAL FOR REDUCED CONSTRUCTION SCHEDULE
- UTILIZATION OF LIGHT WATER REACTOR TECHNOLOGY

ceramic core, low power density and small core diameter makes it possible to remove all decay heat by conduction and radiation from the core.

Feasibility studies⁽²⁷⁾ for a two unit HTGR for cogeneration at a large oil refinery show the concept to be viable. Capital costs for the two unit HTGR were estimated at \$1.264 billion in January 1980 dollars. Presumably the cost of an 800 MWe two unit HTGR station would be comparable. Development costs are not included in the estimate.

Cost estimates for conceptual designs, such as CNSS and HTGR are subject to considerable uncertainty. To obtain a more firm basis for cost estimation would require that a prototype plant be constructed and licensed. The benefits of inherent safety and, therefore, better public and regulatory acceptance could make these or similar designs very attractive.

6.0 EXPORT MARKET FOR SMALL PLANTS

To maintain a stable electrical grid, the maximum single unit generating capacity is generally limited to about ten percent of total capacity. This rule-of-thumb has not always been followed but in many cases the underlying rationale has influenced less developed countries with a small total capacity to opt for small units, in some cases nuclear. An export market for small reactors has existed in the past. Small reactors are currently operating or under construction in a number of countries including Argentina, Brazil, Bulgaria, Finland, Korea, Mexico, Philippines, Romania, Taiwan and Yugoslavia. An export market has also developed for large reactors, and supplier countries are competing vigorously for this market, which includes such countries as Brazil, China, Korea, South Africa, Spain and Taiwan. For U.S. vendors this is the only currently active market. AECL, the Canadian vendor, is currently developing a large version of CANDU to compete in this market.⁽²⁸⁾

6.1 Extent of Nuclear Powerplant Export Market

The current export market for small reactors is envisioned as the less developed countries. While there have been a number of expressions of interest (eg. Turkey, Bangladesh), there has been no significant recent activity in this market. One problem is that the countries who have expressed an interest lack the financial resources to pursue the nuclear option. It appears that this market alone would not be sufficient to justify the development of small reactors. Nevertheless, most supplier countries either offer a small reactor option, or are developing

designs for this market.

The International Atomic Energy Agency (IAEA) has actively promoted what they refer to as Small and Medium-sized Power Reactors (SMPRs). These are defined by IAEA as reactors less than 600 MWe. However, even recent articles by IAEA personnel^(29,30,31) acknowledge the limited nature of the SMPR market with such statements as:

- At any time ten to 15 countries represent an exclusive market for SMPRs; a typical residence time for any one country in this range is about 15 years.
- The prospects are best (that nuclear power can be competitive in developing countries) where a 900 MWe or larger unit can be accommodated.

In the event that small (and presumably less costly) reactors eventually find a place in the market which is now dominated by large reactors, the less developed countries would likely provide a market for some additional units. In this case, the development cost would be spread over a large number of units and the "first-of-a-kind" risk would not be born by the less developed countries. In the event that new less costly small reactors do not enter the world market, the limited market in less developed countries could very likely be met by existing small reactor designs.

6.2 Developments in Supplier Countries

There are a number of small reactors and small reactor designs currently available. Some of these are well established designs which have been in operation for many years. Others are new concepts which have yet to be proven. A renewed interest in small reactors is apparent from the recent higher level of design activity in this area. Following is a very brief summary of available reactors and designs.

6.2.1 Germany

KWU has participated in the small reactor export market with their pressurized heavy water reactor design (Atucha-Argentina). However, there has been considerable recent activity directed at developing and marketing a small (200 MWe) BWR design with several inherent safety features.⁽³²⁾ The design draws heavily upon well proven BWR fuel technology but uses an indirect cycle. The single steam generator utilizes steam flow on the primary side. Inherent safety features include natural circulation cooling and a suppression pool located above the vessel from which passive safety injection is provided. The power density has been lowered to about one half that of present generation large reactors. This has the benefit of allowing a two to three year fuel cycle. Figure 10 is a schematic of the design.

Interatom has also been active in the small reactor development area with a proposed Integrated Pressurized Water Reactor (IPWR) design. This design evolved from the Otto Hahn ship propulsion reactor. An

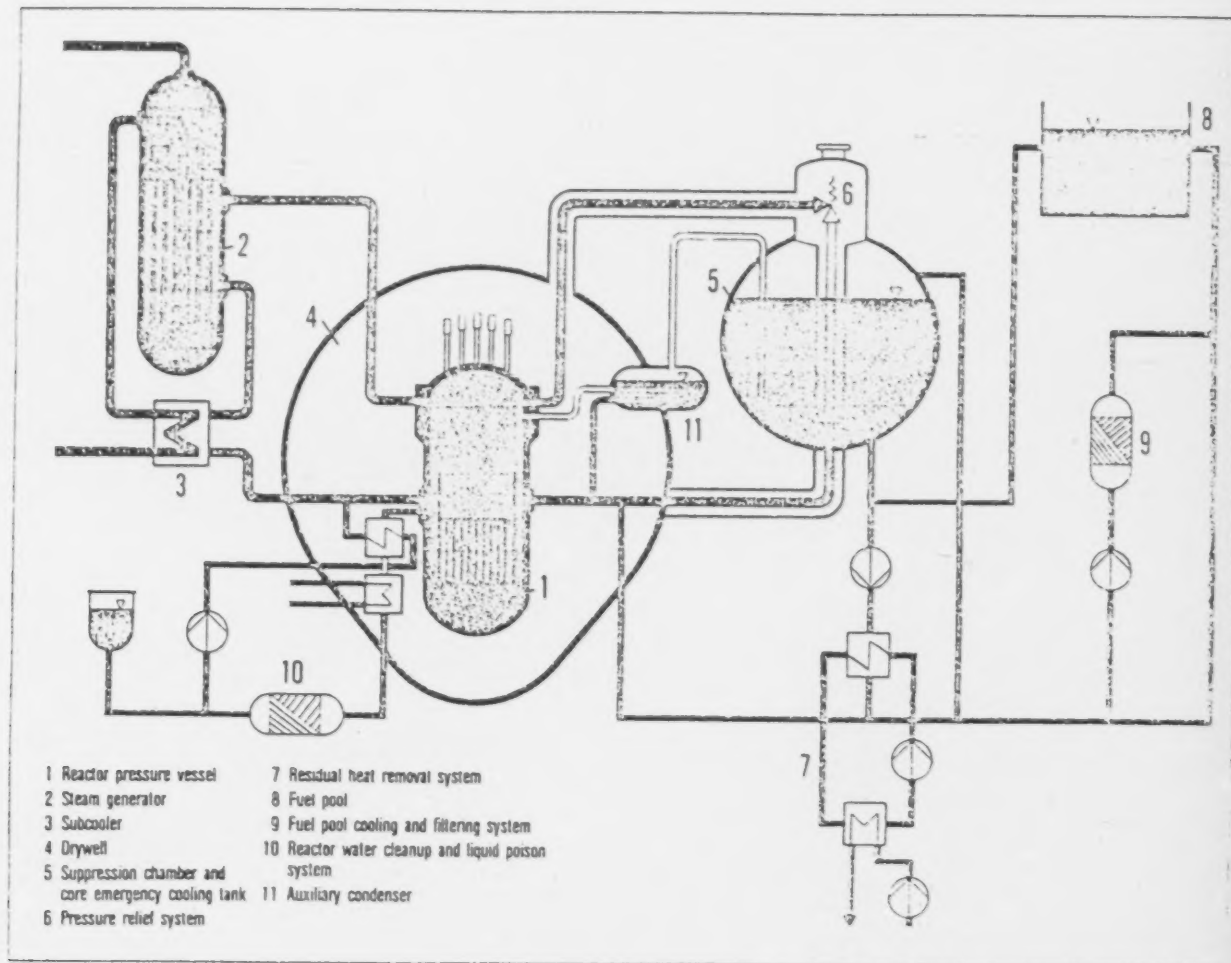


FIGURE 10 KWU SMALL BWR SCHEMATIC

effort on a modular HTGR system is also underway based on experience with the AVR plant.

6.2.2 Japan

The Ministry of International Trade and Industry (MITI) is conducting a feasibility study on small and medium size multi-purpose reactors.⁽³³⁾ Such reactors are being considered for both the domestic and export market. A 200 MWe conventional BWR is also being developed jointly with Hitachi, mainly for export.

Other programs by MITI/Agency of Natural Resources and Energy (ANRE) and JAERI aimed at providing industrial process heat and district heating are discussed in Reference 25.

6.2.3 France

Framatome has designed a two-loop 600 MWe PWR for the export market.⁽³⁴⁾ The NSSS design appears to be similar to the Westinghouse two loop plants. Framatome was involved in design and fabrication of reactor vessels for several of the two loop Westinghouse plants. The design draws heavily on Framatome experience with large plants and uses many of the same components. The plant features a compact design with safety and auxiliary systems located in a ring building which encircles the reactor building.

Technicatome is working on a small PWR design, referred to as NP-300, which is based on ship propulsion reactor technology. Reference 25 has a fairly extensive description of this project. The design

features separate nuclear and conventional islands and a steel cylindrical containment. Two steam generators are directly linked to the reactor vessel by a very short connecting piece which surrounds concentrically the hot and cold legs of the primary system. This absence of primary piping limits the maximum LOCA to the size of small connecting piping, an inherent safety feature. The refueling cycle is two years and the delivery time is said to be around five years.

6.2.4 Sweden

ASEA-ATOM has exported small BWRs to Finland and at one time had received a letter of intent (since withdrawn) from Turkey for a small BWR. However, current interest centers on an inherently safe novel design called SECTUS⁽³⁵⁾ (Safe and Environmentally Clean Thermal Urban Station) which is an outgrowth of the SECURE⁽³⁶⁾ (Safe and Environmentally Clean Urban Reactor).

The entire SECTUS reactor is located within a very large prestressed concrete vessel filled with highly borated water. In normal operation, the borated water is kept out of the reactor loop by maintaining a pressure balance across a honeycomb baffle. The location of the hot/cold interface is controlled by varying the main circulation pump speed to maintain the hydrostatic pressure balance. Should a void be formed in the core, the hydrostatic pressure balance will be upset causing the highly borated water to flow into the main coolant loop, thus shutting down the reaction. The large water inventory within the concrete vessel acts as a heat sink for an extended period of time. Active safety systems are not required.

Other inherent safety/safeguards features include a lower core power density than present generation reactors and a lower operating pressure (1015 psia) than current PWRs. Both the SECTUS and SECURE designs are based upon the same safety philosophy of dependence upon inherent, passive features of the system. This concept is referred to as PIUS (Process Inherent Ultimate Safety) by ASEA-ATOM.

6.2.5 United Kingdom

The British have an interesting history regarding small reactors. The MAGNOX natural uranium-graphite reactor is a small reactor which is the mainstay of the nuclear program. A version of this reactor is reportedly available for export markets. Considerable development was also expended on the Steam Generating Heavy Water Reactor (SGHWR) which is similar to CANDU. However, the present nuclear program includes Sizewell B, a large PWR unit. Any advocacy of small reactors will have to carefully consider the situation in Britain where the current move is from small to large reactors.

One of the recent more interesting developments in Britain is Rolls-Royce and Associates development of a small PWR as an outgrowth of their ship propulsion work. The Rolls-Royce design is a 300 MWe three loop PWR which can be mounted on a barge. The complete power station could be mounted on two to three barges.

6.2.6 Canada

The CANDU reactor has a long and successful history of reliable power production in Canada. Some success was also achieved in the export market. Figure 11 is a schematic of the CANDU reactor.

Reference 37 identifies a number of safety-related characteristics of CANDU, namely:

- presence of a large low-temperature heat sink (the heavy water moderator) dispersed throughout the core.
- two independent physically separate and diverse shutdown systems.
- availability of a shutdown cooling system which is capable of removing decay heat at full pressure.

The overall probability of a large scale core meltdown in CANDU reactors is therefore well below 10^{-6} per reactor-year which is significantly less than WASH-1400 and the FRG Risk Study values for LWRs.

6.2.7 India

India has produced a 220 MWe natural uranium fueled Pressurized Heavy Water Reactor for its own use. While this plant has apparently not been offered for export, this possibility exists in the future.

6.2.8 U.S.S.R.

Atomenergo Export has exported their 440 MWe PWR to Finland and Eastern Block countries. The design features six coolant loops and two turbine generators.⁽³⁸⁾ A sketch of the plant is shown in Figure 12.

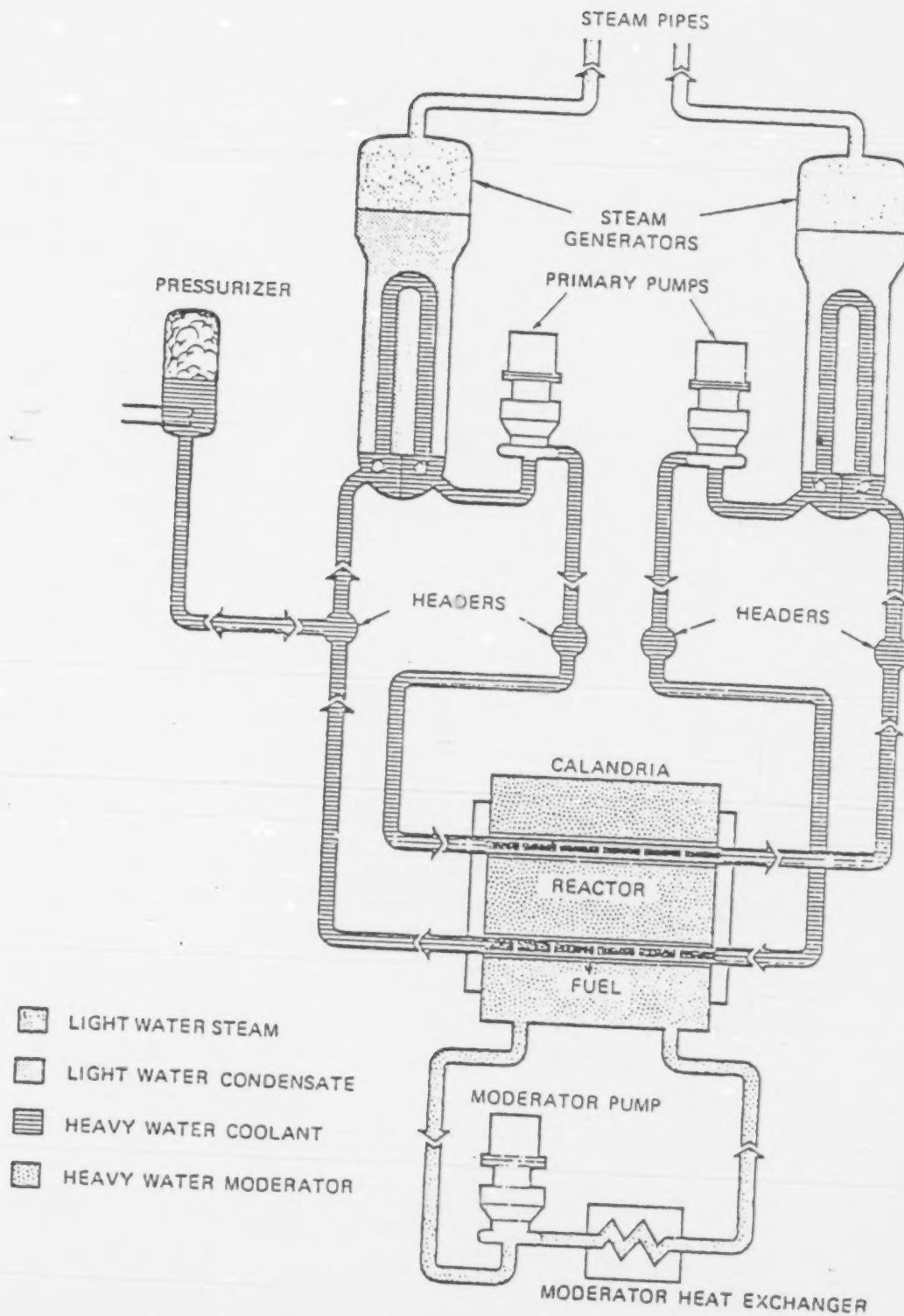


Fig. 11 CANDU Reactor Simplified Flow Diagram

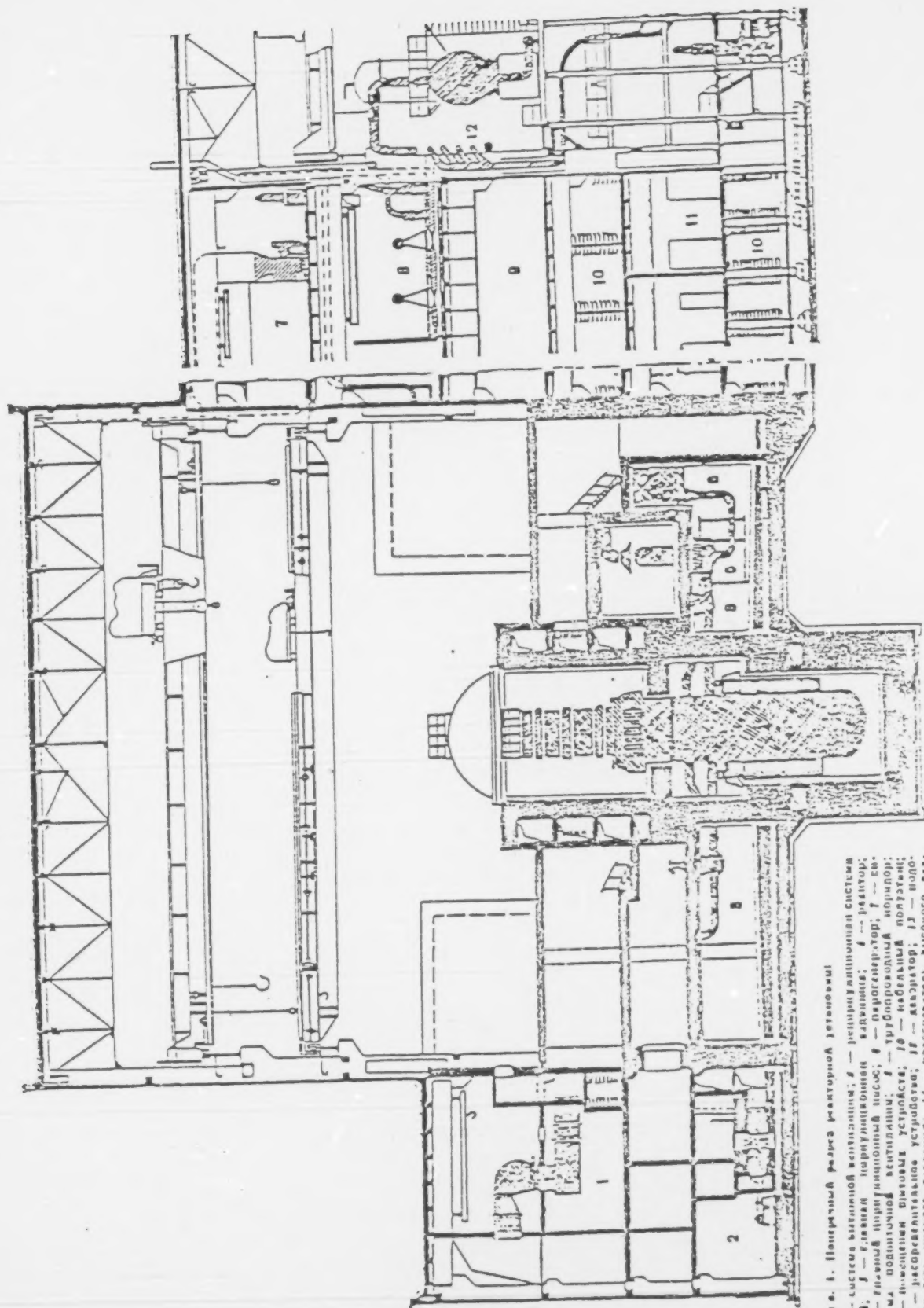


Рис. 1. Принципиальная схема реакторной установки
 1 — система водоподготовки; 2 — реактор; 3 — система водоподготовки; 4 — реактор; 5 — реактор; 6 — реактор; 7 — реактор; 8 — реактор; 9 — реактор; 10 — реактор; 11 — реактор; 12 — реактор; 13 — реактор; 14 — реактор; 15 — реактор; 16 — реактор.

FIGURE 12 USSR 440 MWe PWR PLANT LAYOUT

6.3 Export Potential for U.S. Reactors

The export potential for small U.S. reactors will be determined by a number of factors, including the availability of products in the market. At present, the domestic vendors appear to be concentrating their efforts on the large reactor export market. Other supplier countries are presently developing conceptual designs which are aimed specifically at the small reactor export market, eg. the KWU small BWR design.

In the past, U.S. vendors have obtained a significant share of the export market, principally by virtue of technological and price advantages. Recently these advantages seem to have become less significant as other supplier countries became stronger technologically. At the same time, the legal and financing positions of U.S. vendors which had been about equal to other suppliers have eroded somewhat. Therefore, the U.S. does not appear to enjoy the strong position which it held in the past in the export market. If this is the case, then it is not surprising that U.S. vendors are not actively seeking the small reactor export market.

However, should small reactors be introduced into the domestic market and should reasonable construction costs be achieved, then the potential for additional sales would exist in the export market.

7.0 SUMMARY

Large reactors have traditionally benefited from economies of scale which made their unit cost of generation lower than that of small reactors. However, greatly increased construction costs and lengthening schedules have called the traditional economies of scale into question. Small reactors offer the potential for simplified design, shop fabrication of major subassemblies and shortened construction schedules. Small reactors are therefore the subject of renewed interest.

A review of the technical, safety and economic aspects of reactor size has revealed a number of potential advantages of small reactors.

1. Construction cost increases for large reactors are due to the high cost of field fabrication, significant additional unanticipated engineering manhours after the start of construction and the cost of capital during construction. Small reactors offer the possibility of complete engineering before construction, shop fabrication of major subassemblies and shorter construction schedules. This challenges the traditional economies of scale.
2. Historical availability data show that small reactors are more reliable than large reactors which came on line during the same time period, in the sense that their average availability is higher.
3. A statistical analysis of accident precursors performed by ORNL showed that the frequency of occurrence of severe core damage accident precursors is about the same for small and large plants.

However, based on this same data, it is conjectured that accident initiators are on the average more readily controlled in small plants.

4. Evolutionary design and safety improvements can be incorporated in either small or large plants. However, revolutionary concepts are more easily incorporated in small plants. Some utility operating personnel have expressed the opinion that the present generation of nuclear plants are too large and complex. Many of the preferred characteristics of new LWRs desired by utilities would be easier to incorporate in small reactor designs. Inherent safety features may also be easier to include.

Concerns of intervenors do not appear to depend significantly on reactor size. Rather they depend upon what is perceived as the worst possible accident consequences. While small reactors would not of themselves reduce concerns, they may be more readily designed with inherent safety characteristics which would reduce concerns.

Domestic utilities are not currently ordering new nuclear generating capacity: There is, therefore, no immediate market for plants of any size. However, small plants may offer advantages for future new baseload generation capability since they fit in better with the new lower load growth patterns and utility planning horizons.

Other supplier countries are actively developing new small reactor designs apparently for the export market. This is perceived to be a rather limited market in which the U.S. position, although once very

strong, is now eroded by strong competition and institutional factors. Therefore, development of small reactor designs for this market does not seem justifiable for U.S. suppliers. However, should a domestic market develop for small reactors, the export market would offer the possibility of additional sales.

8.0 REFERENCES

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